RULEMAKING ISSUE
AFFIRMATION

July 24, 2003 SECY-03-0127

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: FINAL RULEMAKING—RISK-INFORMED 10 CFR 50.44, "COMBUSTIBLE GAS CONTROL IN CONTAINMENT"

PURPOSE:

To obtain Commission approval to publish the final rule and the regulatory guidance implementing the rule.

SUMMARY:

This final rule amends NRC's regulations governing the domestic licensing of production and utilization facilities. Specifically, the rule eliminates the requirements for hydrogen recombiners and hydrogen purge systems in currently licensed light water reactors and relaxes the requirements for hydrogen and oxygen monitoring equipment commensurate with the equipment's risk significance. The rule also specifies requirements for combustible gas control in future water-cooled reactors and non-water-cooled reactors.

BACKGROUND:

In SECY-01-0162, "Staff Plans for Proceeding With the Risk-Informed Alternative to the Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors in 10 CFR 50.44," dated August 23, 2001, the staff recommended revising the existing regulations rather than developing a voluntary, risk-informed alternative. In a staff requirements

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301-415-1116

memorandum dated December 31, 2001, the Commission approved the staff's recommendation and requested that the staff explain why installing passive autocatalytic recombiners would not pass a cost benefit test.

Mr. Christie, of Performance Technology, Inc., submitted letters dated October 7 and November 9, 1999, requesting changes to the regulations in § 50.44. The staff has treated Mr. Christie's request as a petition for rulemaking (Docket No. PRM-50-68). The NRC published a notice requesting comment on the petition in the *Federal Register* on January 12, 2000 (65 FR 1829). The staff discussed issues raised by the petitioner in SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)". NRC action on this petition is completed as detailed in the attached *Federal Register* notice. A letter to the petitioner advising him of NRC's final action on his petition will be signed by the Secretary upon approval of the final rule. A copy of this letter is attached (Attachment 1).

The Commission also received a petition for rulemaking from the Nuclear Energy Institute. The petition was docketed on April 12, 2000 (Docket No. PRM-50-71). The staff published a notice requesting comment on the petition in the *Federal Register* on May 30, 2000 (65 FR 34599). The petitioner requested that the NRC amend its regulations in § 50.44 and § 50.46 to allow nuclear power plant licensees to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have been approved by the NRC staff. The petitioner believes the proposed amendment would improve the efficiency of the regulatory process by eliminating the need for licensees to obtain individual exemptions to use advanced cladding materials that have already been approved by the NRC. NRC action on the portion of this petition pertaining to 10 CFR 50.44 is completed as detailed in the attached *Federal Register* notice. A letter to the petitioner advising that the NRC has partially completed action on the petition will be signed by the Secretary upon approval of the final rule. A copy of this letter is attached (Attachment 2). The remaining portion of the petition (relating to 10 CFR 50.46) will be closed out later by separate action.

On May 13, 2002, in SECY-02-0080, "Proposed Rulemaking - Risk-Informed 10 CFR 50.44 - Combustible Gas Control in Containment," the NRC staff provided the Commission with a proposed rulemaking package, including a cost-benefit analysis of installing passive autocatalytic recombiners (in response to a previous Commission request). In a staff requirements memorandum dated June 27, 2002, the Commission approved the issuance of the proposed rule. The NRC then published the rule in the *Federal Register* on August 2, 2002 (67 FR 50374). The public comment period expired on October 16, 2002. The NRC staff has evaluated the comments and has prepared the final rule.

<u>DISCUSSION</u>:

After the 1987 revision of 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," there have been significant advances in our understanding of the risk at nuclear power plants from the production and combustion of hydrogen (and other combustible gases) throughout the spectrum of reactor accidents. These advances are described in SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)."

Final Rule

The final rule retains existing requirements for ensuring a mixed atmosphere, inerting Mark I and Mark II containments, and providing hydrogen control systems capable of accommodating the amount of hydrogen generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments.[1] The rule also retains the existing analysis requirements and equipment survivability requirements for Mark III and ice condenser containments. It eliminates the design basis loss of coolant accident (LOCA) hydrogen release requirement from § 50.44, consolidates the requirements for hydrogen and oxygen monitoring into § 50.44, and relaxes safety classifications and licensee commitments to certain design and qualification criteria. The rule also condenses (without materially changing) the hydrogen control requirements in § 50.34(f) for future water-cooled reactor applicants and licensees and relocates these requirements to § 50.44. It also relocates the high-point vent requirements from § 50.44 to § 50.46a and eliminates a requirement that prohibits venting the reactor coolant system if it could "aggravate the challenge to containment." The final rule also specifies requirements for combustible gas control in future reactors, including evaluation of combustible gases other than hydrogen. The regulatory text in the attached final rule differs from the language published in the proposed rule, reflecting the staff's consideration and resolution of comments made by the Nuclear Energy Institute (NEI) regarding applicability of the rule to future reactor designs. This issue is discussed more fully in the next section. The final rule addresses Mr. Christie's petition and addresses the § 50.44 portion of the NEI petition. The guidance reflects changes made by the final rule, including related changes that allow removal of hydrogen and oxygen monitors from the technical specifications. At plants with inerted containments (Mark I and Mark II BWRs), an existing technical specification for maintaining primary containment oxygen concentration below 4% by volume (i.e., inerted) will be retained.

Public Comments on the Proposed Rule

The Commission received letters from 14 commenters, containing approximately 41 comments on the proposed rule and draft regulatory guide. Seven of the commenters were licensees, two were vendors, two were representatives of utility groups (many of whose members are licensees), two were private citizens, and one represented a citizen group. All comments on the rule and the information in and public comments on the two petitions for rulemaking were considered in formulating the final rule.

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[1]The Feasibility Study, in SECY-00-0198, indicated that some mitigative features may need to be enhanced beyond current requirements. This concern was identified as Generic Issue (GI)189. The resolution of GI-189 will assess whether improvements to safety can be achieved and the costs and benefits of enhancing combustible gas control requirements for Mark III and ice condenser containment designs. The resolution of GI-189 is proceeding independently of this rulemaking. The technical basis for this issue has been evaluated and discussed with the Advisory Committee for Reactor Safeguards. The NRC is establishing requirements to provide backup power to allow the hydrogen igniters already installed in these facilities to function during station blackout events.

Many commenters expressed strong support for the rulemaking to risk-inform the regulations in § 50.44 and commended the NRC for developing a rule based on risk-informed and performance-based insights that will eliminate unnecessary regulatory requirements. One industry commenter stated that this rule will enhance public health and safety because it will increase the reliability of the hydrogen and oxygen monitoring systems. One private citizen questioned why the NRC was considering relaxing requirements that provide protection against some of the uncertainties and hazards of nuclear power. A citizen group opposed the changes, contending that eliminating the design basis accident release and relaxing safety classifications and licensee commitments to certain design and qualification criteria would only benefit the financial interests of licensees. Other comments suggested minor clarifications or editorial changes.

A comment submitted by the Nuclear Energy Institute (NEI) resulted in the most significant modification to language contained in the proposed rule. NEI commented that the proposed rule language in § 50.44(c) was applicable to all future reactors, yet it assumed that future reactors would present the same combustible gas hazard as current light water reactors. NEI recommended that § 50.44(c) be made applicable to light water reactors only. The NRC staff agrees with NEI that the requirements proposed in § 50.44(c) might not be appropriate for some future reactor designs. Thus, the final rule has been changed so that paragraph (c) applies only to water-cooled reactor designs with combustible gas characteristics similar to those of current light water reactors. The NRC has also added a new paragraph (d), that specifies general combustible gas control requirements for non-water-cooled reactors and certain water-cooled reactors with different combustible gas characteristics than current light water reactors. These facilities are required to be designed to tolerate or mitigate effects of any combustible gases generated by design basis or significant beyond-design-basis accidents. A detailed evaluation of all public comments is provided in Section IV of the attached *Federal Register* notice.

Contents of the Final Rulemaking Package

This rulemaking package includes the final rule to be published in the *Federal Register* (Attachment 3) and the final regulatory analysis (Attachment 4). The package also includes the final regulatory guide (Attachment 5) and the revision to the standard review plan (Attachment 6). Technical specification changes associated with the amended regulations will be implemented by the Consolidated Line Item Improvement Process. A model safety evaluation and the associated changes to the standard technical specifications will be published shortly after the final § 50.44 is published.

ACRS and CRGR Reviews

The staff's final rule was reviewed by the Advisory Committee on Reactor Safeguards (ACRS), on April 10, 2003, and by the Committee to Review Generic Requirements (CRGR) in May 2003. Both committees favored issuance of the final rule.

<u>RESOURCES</u>:

The FY 2003 resources to complete and implement the final rulemaking (0.5 FTE for NRR and 0.1 FTE for RES) are included in the FY 2003 budget. The staff does not expect that additional resources will be needed to complete this effort.

COORDINATION:

The Office of the General Counsel has no legal objection to this paper. The Office of the Chief Financial Officer has reviewed this Commission paper for resource implications and has no objections. The ACRS and CRGR have reviewed this final rule.

RECOMMENDATIONS:

That the Commission:

1. *Approve* the notice of final rulemaking for publication (Attachment 3).

2. *Certify* that this rule, if promulgated, will not have a negative economic impact on a substantial number of small entities in order to satisfy requirements of the Regulatory Flexibility Act, 5 U.S.C. 605(b).3.

3 *Note*:

   a. The following documents will be published in the *Federal Register*:

      - The final rule, including the Finding of No Significant Environmental Impact (Attachment 3)
      - Notice of availability of the final regulatory analysis (Attachment 4, also available in Public Document Room and on the NRC rulemaking Web site)
      - Notice of availability of the final Regulatory Guide 1.7, Revision 3, ?Control of Combustible Gas Concentrations in Containment" (Attachment 5)
      - Notice of availability of the revision to Standard Review Plan Section 6.2.5, ?Combustible Gas Control in Containment" (Attachment 6)

   b. The Chief Counsel for Advocacy of the Small Business Administration will be informed of the certification regarding economic impact on small entities and the basis for it, as required by the Regulatory Flexibility Act.

   c. The NRC has determined that this action is not a major rule under the Small Business Regulatory Enforcement Fairness Act of 1996 (SBREFA) and has confirmed this determination with the Office of Management and Budget (OMB).

   d. Copies of the final rule will be distributed to all affected Commission licensees. The notice will be sent to other interested parties upon request.

   e. Letters informing the petitioners of the Commission's decisions on their petitions are attached for the Secretary's signature (Attachments 1 and 2).

f. A public announcement will be issued.

g. The appropriate congressional committees will be informed.

/RA by Patricia Norry Acting For/

William D. Travers
Executive Director
for Operations

Attachments:
1. Letter to Bob Christie
2. Letter to NEI
3. *Federal Register* notice with final rule
4. Regulatory Analysis
5. Regulatory Guide 1.7, Revision 3
6. Standard Review Plan Section 6.2.5

Mr. Bob Christie
Performance Technology
P. O. Box 51663
Knoxville, Tennessee 37950-1663

SUBJECT: FINAL NRC ACTION ON PETITION FOR RULEMAKING PRM-50-68

Dear Mr. Christie:

In your letter to the NRC Commissioners dated October 7, 1999, supplemented by a letter to the NRC staff on November 9, 1999, you filed a petition for rulemaking requesting the Nuclear Regulatory Commission (NRC) to amend its regulations regarding hydrogen control systems at nuclear power plants. You stated that the current regulations for hydrogen control systems at some nuclear power plants are detrimental and present a health risk to the public. You also believe that similar detrimental situations may apply to other systems as well. You stated that your proposed amendments would eliminate those situations associated with hydrogen control systems that present adverse conditions at nuclear power plants. The petition was docketed as PRM-50-68 on November 15, 1999. On January 12, 2000, the NRC published a notice of receipt of this petition in the *Federal Register* (65 FR 1829). The notice summarized the issues in the petition and solicited public comments on those issues.

In your petition, you state that you performed a detailed review of the NRC staff's San Onofre Task Zero Safety Evaluation Report (Pilot Program for Risk-Informed Performance-Based Regulation), dated September 3, 1998, with respect to that plant's hydrogen control system. Based upon this review, you requested the NRC to amend its regulations in the following areas:

1. Retain the existing requirement for inerting the atmosphere of existing Mark I and Mark II containments.

2. Retain the existing requirement for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal-water reaction involving 75% of the fuel cladding.

3. Require all future light water reactors to postulate a 75% metal-water reaction (instead of the 100% required by the current rule) for analyses undertaken pursuant to §50.44(c).

4. Retain the existing requirements for high-point vents.

5. Eliminate the existing requirement in §50.44(b)(2) to insure a mixed atmosphere in containment.

6. Eliminate the existing requirement for hydrogen releases during design basis accidents of an amount equal to that produced by a metal/water reaction of 5% of the cladding.

7. Eliminate the requirement for hydrogen recombiners or purging in light water reactor (LWR) containments.

8. Eliminate the existing requirements for hydrogen and oxygen monitoring in LWR containments.

9. Revise General Design Criterion 41, “Containment Atmosphere Cleanup”, to require systems to control fission products and other substances that may be released into the reactor containment only for accidents where there is a high probability that fission products will be released to the reactor containment.

10. Additionally, you stated that during the San Onofre review, the NRC granted an exemption from the design basis requirements for the hydrogen control system based on information obtained from analysis of severe accidents. You stated that the NRC staff’s evaluation indicated that adherence to the requirements for certain design basis accidents at San Onofre could have a detrimental effect on public health and safety. You believe that there may be other instances at facilities when adherence to design basis accident requirements could be detrimental to safety. Thus, you requested the NRC to issue an interim policy statement applicable to all NRC staff to ensure that the NRC Executive Director for Operations was promptly notified whenever the staff discovered cases when compliance with design basis accident requirements was detrimental to public health.

The Commission considered the issues you raised in your petition and the public comments received in conjunction with an ongoing effort to risk-inform the regulations for combustible gas control. Each of the issues raised in your petition is addressed in a final rule amending 10 CFR 50.44, “Combustible Gas Control in Containment.” The Commission concluded that issues 1, 2, 6, and 7 above will be granted; issue 4 will be granted in part; and issues 3, 5, 8, 9, and 10 will be denied upon promulgation of the amended rules. Subheading V, “Petition for Rulemaking, PRM-50-68," of the enclosed *Federal Register* notice of the final rule contains a discussion of the issues in your petition and the Commission’s resolution of each of those issues.

Sincerely,

Annette Vietti-Cook
Secretary of the Commission

Docket: PRM-50-71

Enclosure: Federal Register Notice

Alex Marion
Director, Engineering
Nuclear Generation Division
Nuclear Energy Institute
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Washington, DC 20006

SUBJECT: FINAL NRC ACTION ON PETITION FOR RULEMAKING PRM-50-71

Dear Mr. Marion:

In a letter dated April 12, 2000, the Nuclear Energy Institute (NEI) filed a petition for rulemaking requesting the Nuclear Regulatory Commission (NRC) to amend its regulations regarding hydrogen control systems at nuclear power plants. The petition was published in the *Federal Register* for public comment on May 31, 2000. The NEI requested that the NRC amend its regulations to allow nuclear power plant licensees to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have been approved by the NRC staff. You believe the proposed amendment will improve the efficiency of the regulatory process by eliminating the need for individual licensees to obtain exemptions to use advanced cladding materials that have already been approved by the NRC.

Specifically, the NEI stated that the NRC's current regulations require uranium oxide fuel pellets used in commercial reactor fuel, to be contained in cladding material made of Zircaloy or ZIRLO. The requirement to use either of these materials is stated in 10 CFR 50.44 and 10 CFR 50.46. You noted that prior to promulgation of these regulations, commercial nuclear fuel vendors have developed and continue to develop materials other than Zircaloy or ZIRLO that NRC reviews and approves for use in commercial power reactor fuel. Each of these approvals requires the NRC to grant an exemption to the licensee who requests to use fuel with these cladding materials. You requested that NRC amend §50.44 and §50.46 of its regulations to allow licensees discretion to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided that the cladding materials meet the fuel cladding performance requirements and have been reviewed and approved by NRC staff. You noted that during the past 9 years there have been at least eight requests for exemptions, that each exemption has cost more than $50,000, and that the requests for exemption have become increasingly more frequent, resulting in inefficiencies. You stated that the proposed amendment would remove an unwarranted licensing burden without increasing risk to public health and safety.

The Commission has evaluated the portion of your petition regarding the changes requested to 10 CFR 50.44 and the associated public comments, and has determined that portion of the petition should be denied. The issue regarding 10 CFR 50.44 is addressed in a final rule amending 10 CFR Part 50.44, "Combustible Gas Control in Containment." The regulation has been amended the so that it does not refer to fuel with specific types of Zircaloy cladding; instead, the rule applies to all boiling and pressurized water reactor fuel. Thus, even though the revised rule does not contain the language changes that you requested, the rule accomplishes your intended purpose with respect to 10 CFR 50.44 by other means. Subheading VI, "Petition for Rulemaking PRM-50-71," of the enclosed *Federal Register* notice contains a detailed discussion of the 10 CFR 50.44 issue in your petition and the Commission's resolution of that issue. The portion of your petition that pertains to changes you requested to 10 CFR 50.46 will be evaluated separately by the Commission at a later time.

Sincerely,

Annette Vietti-Cook
Secretary of the Commission

Docket: PRM-50-71

Enclosure: *Federal Register* Notice

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50 and 52

RIN 3150 - AG76

Combustible Gas Control in Containment

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations for combustible gas control in power reactors applicable to current licensees and is consolidating combustible gas control regulations for future reactor applicants and licensees. The final rule eliminates the requirements for hydrogen recombiners and hydrogen purge systems, and relaxes the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance. This action stems from the NRC's ongoing effort to risk-inform its regulations, and is intended to reduce the regulatory burden on present and future reactor licensees. Additionally, the final rule grants in part and denies in part a petition for rulemaking (PRM-50-68) submitted by Mr. Bob Christie. This notice constitutes final NRC action on PRM-50-68. The final rule also denies part of a petition for rulemaking (PRM-50-71) submitted by the Nuclear Energy Institute. The remaining issue in PRM-50-71 that is not addressed by this final rule will be evaluated in a separate NRC action. The NRC has updated a guidance document, "Control of Combustible Gas Concentrations in Containment" to address changes in the rule. A draft regulatory guide containing the revisions was published for comment with the proposed rule.

EFFECTIVE DATE: (Insert date 30 days after the date of publication).

FOR FURTHER INFORMATION CONTACT: Richard Dudley, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone (301) 415-1116; e-mail: rfd@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Background

On October 27, 1978 (43 FR 50162), the NRC adopted a new rule, 10 CFR 50.44, specifying the standards for combustible gas control systems. The rule required the applicant or licensee to show that during the time period following a postulated loss-of-coolant accident (LOCA), but prior to effective operation of the combustible gas control system, either: (1) An uncontrolled hydrogen-oxygen recombination would not take place in the containment, or (2) the plant could withstand the consequences of an uncontrolled hydrogen-oxygen recombination without loss of safety function. If neither of these conditions could be shown, the rule required that the containment be provided with an inerted atmosphere to provide protection against hydrogen burning and explosion. The rule defined a release of hydrogen involving up to 5 percent oxidation of the fuel cladding as the amount of hydrogen to be assumed in determining compliance with the rule's provisions. This design-basis hydrogen release was based on the design-basis LOCA postulated by 10 CFR 50.46 and was multiplied by a factor of five for added conservatism to address possible further degradation of emergency core cooling.

The accident at Three Mile Island, Unit 2 involved oxidation of approximately 45 percent of the fuel cladding [NUREG/CR-6197, dated March 1994] with hydrogen generation well in excess of the amounts required to be considered for design purposes by § 50.44. Subsequently, the NRC reevaluated the adequacy of the regulations related to hydrogen control to provide greater protection in the event of accidents more severe than design-basis LOCAs. The NRC reassessed the vulnerability of various containment designs to hydrogen burning, which resulted in additional hydrogen control requirements adopted as amendments to § 50.44. The 1981 amendment, which added paragraphs (c)(3)(i), (c)(3)(ii), and (c)(3)(iii) to the rule, imposed the following requirements:

(1) An inerted atmosphere for boiling water reactor (BWR) Mark I and Mark II containments,
(2) installation of recombiners for light water reactors that rely on a purge or repressurization system as a primary means of controlling combustible gases following a LOCA, and
(3) installation of high point vents to relieve noncondensible gases from the reactor vessel (46 FR 58484; December 2, 1981).

On January 25, 1985 (50 FR 3498), the NRC published another amendment to § 50.44. This amendment, which added paragraph (c)(3)(iv), required a hydrogen control system justified by a suitable program of experiment and analysis for BWRs with Mark III containments and pressurized water reactors (PWRs) with ice condenser containments. In addition, plants with these containment designs must have systems and components to establish and maintain safe shutdown and containment integrity. These systems must be able to function in an environment after burning and detonation of hydrogen unless it is shown that these events are unlikely to occur. The control system must handle an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region.

When § 50.44 was amended in 1985, the NRC recognized that an improved understanding of the behavior of accidents involving severe core damage was needed. During the 1980s and 1990s, the NRC sponsored a severe accident research program to improve the understanding of core melt phenomena, combustible gas generation, transport and combustion, and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (e.g., NUREG-1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light water reactors.

The result of these studies has been an improved understanding of combustible gas behavior during severe accidents and confirmation that the hydrogen release postulated from a design-basis LOCA was not risk-significant because it was not large enough to lead to early containment failure, and that the risk associated with hydrogen combustion was from beyond design-basis (e.g., severe) accidents. These studies also confirmed the assessment of vulnerabilities that went into the 1981 and 1985 amendments that required additional hydrogen control measures for some containment designs.

II. Rulemaking Initiation

In a June 8, 1999, Staff Requirements Memorandum (SRM) on SECY-98-300, Options for Risk-informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities," the NRC approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50. The NRC staff provided its plan and schedule for the study phase of its work to risk-inform the technical requirements of 10 CFR Part 50 in SECY-99-264, "Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50," dated November 8, 1999. The NRC approved proceeding with the plan for risk-informing the Part 50 technical requirements in a February 3, 2000, SRM. Section 50.44 was selected as a test case for piloting the process of risk-informing 10 CFR Part 50 in SECY-00-0086, "Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3)."

Mr. Christie of Performance Technology, Inc. submitted letters, dated October 7 and November 9, 1999, that requested changes to the regulations in § 50.44. He requested that the regulations be amended to:

1. Retain the existing requirement in § 50.44(b)(2)(i) for inerting the atmosphere of existing Mark I and Mark II containments.

2. Retain the existing requirement in § 50.44(b)(2)(ii) for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding.

3. Require all future light water reactors to postulate a 75 percent metal/water reaction (instead of the 100 percent required by the current rule) for analyses undertaken pursuant to

§ 50.44(c).

4. Retain the existing requirements in § 50.44 for high point vents.

5. Eliminate the existing requirement in § 50.44(b)(2) to insure a mixed atmosphere in containment.

6. Eliminate the existing requirement for hydrogen releases during design basis accidents of an amount equal to that produced by a metal/water reaction of 5 percent of the cladding.

7. Eliminate the requirement for hydrogen recombiners or purge in LWR containments.

8. Eliminate the existing requirements for hydrogen and oxygen monitoring in LWR containments.

9. Revise GDC 41 -- Containment Atmosphere Cleanup -- to require systems to control fission products and other substances that may be released into the reactor containment for accidents only where there is a high probability that fission products will be released to the reactor containment.

These letters have been treated by the NRC as a petition for rulemaking and assigned the Docket No. PRM-50-68. The NRC published a document requesting comment on the petition in the Federal Register on January 12, 2000 (65 FR 1829). The issues associated with § 50.44 raised by the petitioner were discussed in SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)." The final rule and the petition are consistent in many areas, but differ regarding the functional requirements for hydrogen and oxygen monitoring, the requirement for ensuring a mixed atmosphere, the source term of hydrogen for water-cooled reactors to analyze in order to ensure containment integrity, and the need to revise GDC-41. The NRC's detailed basis for including these requirements in the rule is addressed in a subsequent section of this supplementary information.

The NRC also received a petition for rulemaking filed by the Nuclear Energy Institute. The petition was docketed on April 12, 2000, and has been assigned Docket No. PRM-50-71. The petitioner requests that the NRC amend its regulations to allow nuclear power plant licensees to use zirconium-based cladding materials other than zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have received approval by the NRC staff. The petitioner believes the proposed amendment would improve the efficiency of the regulatory process by eliminating the need for individual licensees to obtain exemptions to use advanced cladding materials that have already been approved by the NRC. The change would remove the language in 10 CFR 50.44 regarding the use of zirconium-based cladding materials other than Zircaloy or ZIRLO. The NRC published a document requesting comment on the petition in the Federal Register on May 30, 2000 (65 FR 34599). The requested change is unrelated to the risk-informing of 10 CFR 50.44. The NRC addressed the NEI petition in this rulemaking for effective use of resources. Although the final rule does not contain the rule language changes requested by the petitioner, in its revision to 10 CFR 50.44, the NRC eliminated the old language referring to various types of fuel cladding. Thus, the final rule resolves the petitioner's concern regarding §50.44. The NRC's detailed basis for this decision is addressed in a subsequent section of this supplementary information.

In SECY-00-0198, dated September 14, 2000, the NRC staff proposed a risk-informed voluntary alternative to the current § 50.44. Attachment 2 to that paper, hereafter referred to as the Feasibility Study, used the framework described in Attachment 1 to the paper and risk insights from NUREG-1150 and the IPE programs to evaluate the requirements in § 50.44. The Feasibility Study found that combustible gas generated from design-basis accidents was not risk-significant for any containment type, given intrinsic design capabilities or installed mitigative features. The Feasibility Study also concluded that combustible gas generated from severe accidents was not risk significant for: (1) Mark I and II containments, provided that the required inerted atmosphere was maintained; (2) Mark III and ice condenser containments, provided that the required igniter systems were maintained and operational, and (3) large, dry and sub-atmospheric containments because the large volumes, high failure pressures, and likelihood of random ignition to help prevent the build-up of detonable hydrogen concentrations.

The Feasibility Study did conclude that the above requirements for combustible gas mitigative features were risk-significant and must be retained. Additionally, the Feasibility Study also indicated that some mitigative features may need to be enhanced beyond current requirements. This concern was identified as Generic Safety Issue-189 (GI-189). The resolution of GI-189 will assess the costs and benefits of improvements to safety which can be achieved by enhancing combustible gas control requirements for Mark III and ice condenser containment designs. The resolution of GI-189 is proceeding independently of this rulemaking. In an SRM dated January 19, 2001, the NRC directed the NRC staff to proceed expeditiously with rulemaking on the risk-informed alternative to § 50.44.

In SECY-01-0162, "Staff Plans for Proceeding with the Risk-Informed Alternative to the Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors in 10 CFR 50.44," dated August 23, 2001, the NRC staff recommended a revised approach to the rulemaking effort. This revised approach recognized that risk-informing Part 50, Option 3 was based on a realistic reevaluation of the basis of a regulation and the application of realistic risk analyses to determine the need for and relative value of regulations that address a design-basis issue. The result of this process necessitates a fundamental reevaluation or "rebaselining" of the existing regulation, rather than the development of a voluntary alternative approach to rulemaking. On November 14, 2001, in response to NRC direction in an SRM dated August 2, 2001, the NRC staff published draft rule language on the NRC web site for stakeholder review and comment. In an SRM dated December 31, 2001, the NRC directed the staff to proceed with the revision to the existing § 50.44 regulations.

III. Final Action

The NRC is retaining existing requirements for ensuring a mixed atmosphere, inerting Mark I and II containments, and hydrogen control systems capable of accommodating an amount of hydrogen generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments. The NRC is eliminating the design-basis LOCA hydrogen release from § 50.44 and consolidating the requirements for hydrogen and oxygen monitoring into § 50.44 while relaxing safety classifications and licensee commitments to certain design and qualification criteria. The NRC is also relocating and rewording without materially changing the hydrogen control requirements in § 50.34(f) to § 50.44. The high point vent requirements are being relocated from § 50.44 to a new § 50.46a with a change that eliminates a requirement prohibiting venting the reactor coolant system if it could “aggravate” the challenge to containment.

Substantive issues are addressed in the following sections.

A. Retention of Inerting, BWR Mark III and PWR Ice Condenser Hydrogen Control Systems, Mixed Atmosphere Requirements, And Associated Analysis Requirements

The final rule retains the existing requirement in § 50.44(c)(3)(i) to inert Mark I and II type containments. Given the relatively small volume and large zirconium inventory, these containments, without inerting, would have a high likelihood of failure from hydrogen combustion due to the potentially large concentration of hydrogen that a severe accident could cause.

Retaining the requirement maintains the current level of public protection, as discussed in Section 4.3.2 of the Feasibility Study.

The final rule retains the existing requirements in § 50.44(c)(3)(iv), (v), and (vi) that BWRs with Mark III containments and PWRs with ice condenser containments provide a hydrogen control system justified by a suitable program of experiment and analysis. The amount of hydrogen to be considered is that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). The analyses must demonstrate that the structures, systems and components necessary for safe shutdown and maintaining containment integrity will perform their functions during and after exposure to the conditions created by the burning hydrogen. Environmental conditions caused by local detonations of hydrogen must be included, unless such detonations can be shown unlikely to occur. A significant beyond design-basis accident generating significant amounts of hydrogen (on the order of Three Mile Island, Unit 2, accident or a metal water reaction involving 75 percent of fuel cladding surrounding the active fuel region) would pose a severe threat to the integrity of these containment types in the absence of the installed igniter systems. Section 4.3.3 of the Feasibility Study concluded that hydrogen combustion is not risk-significant, in terms of the framework document’s quantitative guidelines, when igniter systems installed to meet § 50.44(c)(3)(iv), (v), and (vi) are available and operable. The NRC retains these requirements. Previously reviewed and approved licensee analyses to meet the existing regulations constitute compliance with this section. The results of these analyses must continue to be documented in the plant’s Updated Final Safety Analysis Report in accordance with § 50.71(e).

The final rule also retains the § 50.44(b)(2) requirement that containments for all currently-licensed nuclear power plants ensure a mixed atmosphere. A mixed containment atmosphere prevents local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment.

B. Elimination of Design-Basis LOCA Hydrogen Release

The final rule removes the existing definition of a design-basis LOCA hydrogen release and eliminates requirements for hydrogen control systems to mitigate such a release at currently-licensed nuclear power plants. The installation of recombiners and/or vent and purge systems previously required by § 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The NRC finds that this hydrogen release is not risk-significant. This finding is based on the Feasibility Study which found that the design-basis LOCA hydrogen release did not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. The requirements for combustible gas control that were developed after the Three Mile Island Unit 2 accident were intended to minimize potential additional challenges to containment due to long term residual or radiolytically-generated hydrogen. The NRC found that containment loadings associated with long term hydrogen concentrations are no worse than those considered in the first 24 hours and therefore, are not risk-significant. The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by licensee implementation of the severe accident management guidelines (SAMGs) or other ad hoc actions because of the long period of time available to take such action. Therefore, the NRC eliminates the hydrogen release associated with a design-basis LOCA from § 50.44 and the associated requirements that necessitated the need for the hydrogen recombiners and the backup hydrogen vent and purge systems.

In plants with Mark I and II containments, the containment atmosphere is required to be maintained with a low concentration of oxygen, rendering it inert to combustion. Mark I and II containments can be challenged beyond 24 hours by the long-term generation of oxygen through radiolysis. The regulatory analysis for this proposed rulemaking found the cost of maintaining the recombiners exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late time frame. The NRC believes that this conclusion would also be true for the backup hydrogen purge system even though the cost of the hydrogen purge system would be much lower because the system also is needed to inert the containment.

The NRC continues to view severe accident management guidelines as an important part of the severe accident closure process. Severe accident management guidelines are part of a voluntary industry initiative to address accidents beyond the design basis and emergency operating instructions. In November 1994, current nuclear power plant licensees committed to implement severe accident management at their plants by December 31, 1998, using the guidance contained in NEI 91-04, Revision 1, “Severe Accident Issue Closure Guidelines.” Generic severe accident management guidelines developed by each nuclear steam system supplier owners group includes either purging and venting or venting the containment to address combustible gas control. On the basis of the industry-wide commitment, the NRC is not requiring such capabilities, but continues to view purging and/or controlled venting of all containment types to be an important combustible gas control strategy that should be considered in a plant’s severe accident management guidelines.

C. Oxygen Monitoring Requirements

The final rule amends § 50.44 to codify the existing regulatory practice of monitoring oxygen in currently-licensed nuclear power plant containments that use an inerted atmosphere for combustible gas control. Standard technical specifications and licensee technical specifications currently require oxygen monitoring to verify the inerted condition in containment.

Combustible gases produced by beyond design-basis accidents involving both fuel-cladding oxidation and core-concrete interaction would be risk-significant for plants with Mark I and II containments if not for the inerted containment atmosphere. If an inerted containment was to become de-inerted during a significant beyond design-basis accident, then other severe accident management strategies, such as purging and venting, would need to be considered. The oxygen monitoring is needed to implement these severe accident management strategies, in plant emergency operating procedures, and as an input in emergency response decision making.

The final rule reclassifies oxygen monitors as non safety-related components. Currently, as recommended by the NRC's Regulatory Guide (RG) 1.97, oxygen monitors are classified as Category 1. Category 1 is defined as applying to instrumentation designed for monitoring variables that most directly indicate the accomplishment of a safety function for design-basis events. By eliminating the design-basis LOCA hydrogen release, the oxygen monitors are no longer required to mitigate design-basis accidents. The NRC finds that Category 2, defined in RG 1.97, as applying to instrumentation designated for indicating system operating status, to be the more appropriate categorization for the oxygen monitors, because the monitors will still continue to be required to verify the status of the inerted containment. Further, the NRC believes that sufficient reliability of oxygen monitoring, commensurate with its risk-significance, will be achieved by the guidance associated with the Category 2 classification. Because of the various regulatory means, such as orders, that were used to implement post-TMI requirements, this relaxation may require a license amendment at some facilities. Licensees would also need to update their final safety analysis report to reflect the new classification and RG 1.97 categorization of the monitors in accordance with 10 CFR 50.71(e).

D. Hydrogen Monitoring Requirements

The final rule maintains the existing requirement in § 50.44(b)(1) for monitoring hydrogen in the containment atmosphere for all currently-licensed nuclear power plants. Section 50.44(b)(1), standard technical specifications and licensee technical specifications currently contain requirements for monitoring hydrogen, including operability and surveillance requirements for the monitoring systems. Licensees have made commitments to comply with design and qualification criteria for hydrogen monitors specified in NUREG-0737, Item II.F.1, Attachment 6 and in RG 1.97. The hydrogen monitors are required to assess the degree of core damage during a beyond design-basis accident and confirm that random or deliberate ignition has taken place. Hydrogen monitors are also used, in conjunction with oxygen monitors in inerted containments, to guide response to emergency operating procedures. Hydrogen monitors are also used in emergency operating procedures of BWR Mark III facilities. If an explosive mixture that could threaten containment integrity exists, then other severe accident management strategies, such as purging and/or venting, would need to be considered. The hydrogen monitors are needed to implement these severe accident management strategies.

The final rule reclassifies the hydrogen monitors as non safety-related components for currently-licensed nuclear power plants. With the elimination of the design-basis LOCA hydrogen release (see Item B. earlier), the hydrogen monitors are no longer required to support mitigation of design-basis accidents. Therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in § 50.2. This is consistent with the NRC's determination that oxygen monitors that are used for beyond-design basis accidents need not be safety grade.

Currently, RG 1.97 recommends classifying the hydrogen monitors in Category 1, defined as applying to instrumentation designed for monitoring key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. Because the hydrogen monitors no longer meet the definition of Category 1 in RG 1.97, the NRC believes that licensees' current commitments are unnecessarily burdensome. The NRC believes that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of significant beyond design-basis accidents. Category 3 applies to high-quality, off-the-shelf backup and diagnostic instrumentation. As with the revision to oxygen monitoring, this relaxation may also require a license amendment at some facilities. Licensees will also need to update their final safety analysis report to reflect the new classification and RG 1.97 categorization of the monitors in accordance with 10 CFR 50.71(e).

E. Technical Specifications for Hydrogen and Oxygen Monitors

As discussed in III.C and III.D above, the amended rule requires equipment for monitoring hydrogen in all containments and for monitoring oxygen in containments that use an inerted atmosphere. The rule also requires that this equipment must be functional, reliable, and capable of continuously measuring the concentration of oxygen and/or hydrogen in containment atmosphere following a beyond design-basis accident for combustible gas control and severe accident management, including emergency planning. Because of the importance of these monitors for the management of severe accidents, the staff evaluated whether operability and surveillance requirements for these monitors should be included in the technical specifications.

In order to be retained in the technical specifications, the monitors must meet one of the four criteria set forth by 10 CFR 50.36. These criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As stated in the *Federal Register* notice (60 FR 36953) for the final rule for technical specifications, these criteria were established to address a "trend toward including in technical specifications not only those requirements derived from the analyses and evaluations included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power plants. This extensive use of technical specifications is due in part to a lack of well-defined criteria (in either the body of the rule or in some other regulatory document) for what should be included in technical specifications." As such, the NRC has decided, and established by rule, not to duplicate regulatory requirements in the technical specifications.

Hydrogen and oxygen monitors do not meet criteria 1, 2, or 3 of 10 CFR 50.36 described above. In addition, the Feasibility Study performed by the NRC, and documented in section 4 of Attachment 2 of SECY-00-0198, concluded that the requirement to provide a system to measure the hydrogen concentration in containment does not contribute to the risk estimates for core melt accidents for large dry containments; is not risk significant during the early stages of core melt accidents for Mark I and Mark II containments; and is not risk significant in terms of dealing with the combustion threat of a core melt accident (except for those conditions where the igniters are not operable, e.g., Station Blackout) for Mark III and ice condenser containments.

These conclusions were based on the assumptions that Mark I and Mark II containments are inert and hydrogen igniters are operable for Mark III and ice condenser containments. It should be noted that the existing technical specification requirements for hydrogen igniters and for maintaining primary containment oxygen concentration below 4 percent by volume (i.e., inerted), are not being removed; therefore, the conclusions in the Feasibility Study on the risk significance of the hydrogen monitors remain valid. On this basis, the NRC has concluded that hydrogen monitors do not meet criterion 4 of 10 CFR 50.36.

Oxygen monitoring is not the primary means of indicating a significant abnormal degradation of the reactor coolant pressure boundary. Oxygen monitors are used to determine the primary containment oxygen concentration in boiling water reactors. As stated above, the limit for primary containment oxygen concentration for Mark I and II containments will remain in technical specifications; therefore, a technical specification requirement for oxygen monitors would be redundant. In addition, technical specifications for hydrogen igniters for Mark III containments will remain. The oxygen monitors have been shown by probabilistic risk assessment to not be risk-significant. On this basis, the NRC has concluded that oxygen monitors do not meet criterion 4 of 10 CFR 50.36.

The NRC has several precedents regarding not duplicating regulatory requirements for severe accidents in the technical specifications. The Anticipated Transients Without Scram (ATWS) rule, (10 CFR 50.62) requires each pressurized water reactor to have equipment from sensor output to final actuation device, diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment is required to be designed to perform its function in a reliable manner and has no associated requirements incorporated in the technical specifications. The Station Blackout (SBO) rule, (10 CFR 50.63) requires that each light water reactor must be able to withstand and/or recover from a station blackout event. Section 50.63 also states that an alternate ac power source will constitute acceptable capability to withstand station blackout provided an analysis is performed that demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source and required shutdown equipment are started and lined up to operate. Again, no requirements for the alternate ac source are required to be in technical specifications.

NRC experience with implementation of the above regulations for non safety-related equipment has shown that reliability commensurate with severe accident assumptions is assured without including such equipment in technical specifications. According to the "Final Report - Regulatory Effectiveness of the Station Blackout Rule" (ADAMS ACCESSION NUMBER: ML003741781), the reliability of the alternate ac power source has improved since implementation of the SBO rule. It states:

"Before the SBO rule was issued, only 11 of 78 plants surveyed had a formal EDG reliability program, 11 of 78 plants had a unit average EDG reliability less that 0.95, and 2 of 78 had a unit average EDG reliability of less that 0.90. Since the SBO rule was issued, all plants have established an EDG reliability program that has improved EDG reliability. A report shows that only 3 of 102 operating plants have a unit average EDG reliability less than 0.95 and above 0.90 considering actual performance on demand, and maintenance (and testing) out of service (MOOS) with the reactor at power."

The staff has, therefore, concluded that requirements for hydrogen and oxygen monitors can be removed from technical specifications. The basis for this conclusion is:

1. these monitors do not meet the criteria of 10 CFR 50.36,

2. the amended 10 CFR 50.44 requires hydrogen and oxygen monitors to be maintained reliable and functional, and

3. the regulatory precedents set by the treatment of other equipment for severe accidents required by 10 CFR 50.62 and 50.63.

F. Combustible Gas Control Requirements for Future Applicants

Paragraph (c) of the final rule sets forth combustible gas control requirements in § 50.44 for all future water-cooled nuclear power reactor designs with characteristics (e.g. type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to currently-licensed light-water reactor designs. The NRC's requirements for future reactors previously specified in §50.34(f)(2)(ix) have been reworded for conciseness but without material change and relocated to §50.44(c)(2) to consolidate the combustible gas control requirements in §50.44 for easier reference. This sub-paragraph requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel clad metal-water reaction and must be capable of precluding uniformly distributed concentrations of hydrogen from exceeding 10 percent (by volume). If these conditions cannot be satisfied, an inerted atmosphere must be provided within the containment. The requirements specified today in §50.44(c)(2) are applicable to future water-cooled reactors with the same potential for the production of combustible gas as currently-licensed light-water reactor designs and are consistent with the criteria currently contained in § 50.34(f)(2)(ix) to preclude local concentrations of hydrogen collecting in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate accident mitigating features. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) include the lessening of pressure and temperature loadings on the containment and essential equipment. These requirements reflect the Commission's expectation that future designs will achieve a higher standard of severe accident performance (50 FR 32138; August 8, 1985).

Paragraph 50.44(d) applies to non-water-cooled reactors and water-cooled reactors which have different characteristics regarding the production of combustible gases from current light-water reactors. Because the specific details of the designs and construction materials used in such future reactors cannot now be known, paragraph (d) specifies a general performance-based requirement that future applicants submit information to the NRC indicating how the safety impacts of combustible gases generated during design-basis and significant beyond design-basis accidents are addressed to ensure adequate protection of public health and safety and common defense and security. This information must be based in part upon a design-specific probabilistic risk assessment. The Commission has endorsed the use of PRAs as a tool in regulatory decisionmaking, see *Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement* (60 FR 42622, August 16, 1995), and is currently using PRAs as one element in evaluating proposed changes to licensing bases for currently licensed nuclear power plants, see Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisionmaking: General Guidance* (July 1998) and Standard Review Plan, Chapter 19, “Use of Probabilistic Risk Assessment in Plant-Specific, Risk Informed Decisionmaking: General Guidance,” NUREG-0800 (July 1998). The use of PRA methodologies in determining whether severe accidents involving combustible gas must be addressed by future non-water-cooled reactor designs (and water-cooled designs which have different combustible gas generation characteristics as compared with the current fleet of light-water-cooled reactors) is a logical extension of the NRC’s efforts to expand the use of PRAs in regulatory decisionmaking.

At this time, the NRC is not able to set forth a detailed description of, or specific criteria for defining a “significant” beyond design-basis accident for these future reactor designs, because the fuel and vessel design, cladding material, coolant type, and containment strategy for these reactor designs are unknown at the time of this final rulemaking. Based in part upon the design-specific PRA, the NRC will determine: (i) what type of accident is considered “significant” for each future reactor design, (ii) whether combustible gas control measures are necessary, and if so, (iii) whether the combustible gas control measures proposed for each design provide adequate protection to public health and safety and common defense and security. Although it is impossible at this time to provide a detailed description or criteria for determining what constitutes a "significant" beyond design-basis accident for the future reactors that are subject to this provision, the NRC nonetheless believes that the concept of "significant" with respect to severe accidents has regulatory precedent which will guide the NRC staff's evaluation of the PRA information for future plants. Section 50.34(f)(2)(ix) of the NRC's current regulations already defines what is in essence the significant beyond design-basis accident which future reactor designs comparable to current light-water reactor designs must be capable of addressing, *viz*., an accident comparable to a degraded core accident at a current light-water reactor in which a metal-water reaction occurs involving 100 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). With respect to other "beyond design-basis" accidents, the Commission has addressed anticipated transients without scram (ATWS), and station blackout, which are currently regarded as "beyond design-basis accidents." The nuclear power industry, at the behest of the NRC, has developed severe accident management guidelines to provide for a systematized approach for responding to severe accidents during operations. Finally, the Commission has required all nuclear power plant licensees to implement emergency preparedness planning to address the potential for offsite releases of radiation in excess of 10 CFR Part 100 limits. A careful review of these regulatory efforts discloses a common thread: regulatory actions addressing "beyond design-basis" accidents have generally been determined based upon a consideration of probability of the accident, together with consideration of the potential scope and seriousness of the health and property value impacts to the general public. Thus, it is possible to set forth a high-level conceptual description of a "significant" beyond design-basis accident involving combustible gas for which the Commission intends for future non-water-cooled reactor designers to address. First, such an accident would have relatively low probability of occurrence, based upon the PRA, but would not be so small that the accident would be deemed incredible. Second, a "significant" beyond design-basis accident involving combustible gas would have serious offsite consequences for the public, involving the potential for death or significant acute or chronic health effects to the general public and/or significant radioactive contamination of offsite property which could result in permanent or long-term commitment of property to nuclear use. Such accidents would typically call for activation of offsite emergency preparedness measures in order to mitigate the adverse effects on public health and safety.

The NRC is currently preparing a Draft Regulatory Guide DG-1122 for public comment, in which the terms, “significant sequences” and “significant contributors” are expected to be addressed. In addition, as part of the proposed rulemaking for risk-informing 10 CFR § 50.46 the Commission has instructed the NRC staff to develop suitable metrics for determining the appropriate risk cutoff for defining the maximum LOCA size. The metrics are to take into account the uncertainties inherent in development of PRAs. The NRC expects that these regulatory activities will ultimately result in more detailed examples of the “significant beyond design-basis” concept to assist a potential applicant in developing the design for a future non-water-cooled reactor (and water-cooled reactor designs which are significantly different in concept from current light-water-cooled reactors), and to guide the NRC’s review of an application involving such a design.

G. Clarification and Relocation of High Point Vent Requirements From 10 CFR 50.44 to 10 CFR 50.46a

The final rule removes the current requirements for high point vents from §50.44 and transfers them to a new §50.46a. The NRC is relocating these requirements because high point vents are relevant to emergency core cooling system (ECCS) performance during severe accidents, and the final §50.44 does not address ECCS performance. The requirement to install high point vents was adopted in the 1981 amendment to §50.44. This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.

The final rule amends the language in § 50.44(c)(3)(iii) by deleting the statement, “the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident.” For certain severe accident sequences, the use of reactor coolant system high point vents is intended to reduce the amount of core damage by providing an opportunity to restore reactor core cooling. Although the release of noncondensible and combustible gases from the reactor coolant system will, in the short term, “aggravate” the challenge to containment, the use of these vents will positively affect the overall course of the accident. The release of any combustible gases from the reactor coolant system has been considered in the containment design and mitigative features that are required for combustible gas control. Any reactor coolant system venting is highly unlikely to affect containment integrity; however, such venting will reduce the likelihood of further core damage. Because overall plant safety is increased by venting through high point vents, the final rule does not include this statement in § 50.46a.

H. Elimination of Post-Accident Inerting

The final rule no longer provides an option to use post-accident inerting as a means of combustible gas control. Although post-accident inerting systems were permitted as a possible alternative for mitigating combustible gas concerns after the accident at Three Mile Island, Unit 2, no licensee has implemented such a system to date. Concerns with a post-accident inerting system include increase in containment pressure with use, limitations on emergency response personnel access, and cost. Sections 50.44(c)(3)(iv)(D) and 50.34(f)(ix)(D) of the former rule were adopted to address these concerns. On November 14, 2001, draft rule language was made available to elicit comment from interested stakeholders. The draft rule language recommended eliminating the option to use post-accident inerting as a means of combustible gas control and asked stakeholders if there was a need to retain these requirements. Stakeholder feedback supported elimination of the post-accident inerting option and indicated that licensees do not intend to convert existing plants to use post-accident inerting. Because there is no need for the regulations to support an approach that is unlikely to be used, the NRC has decided to eliminate post-accident inerting requirements in the final rule.

IV. Comments and resolution on proposed rule and draft regulatory guide

The 60-day comment period for the proposed rule closed on October 16, 2002. The NRC received 14 letters, from 14 commenters, containing approximately 43 comments on the proposed rule and draft regulatory guide. Seven of the commenters were licensees, two were vendors, two were representatives of utility groups (the Nuclear Energy Institute and the Nuclear Utility Group on Equipment Qualification), two were private citizens, and one was a citizen group, Nuclear Information and Resource Service. All comments were considered in formulating the final rule. Copies of the letters are available for public inspection and copying for a fee at the Commission's Public Document Room, located at 11555 Rockville Pike, Room O-1 F23, Rockville, Maryland 20852.

Documents created or received at the NRC after October 16, 2002, are also available electronically at the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/reading-rm.html. From this site, the public can gain entry into the NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These same documents also may be viewed and downloaded electronically via the interactive rulemaking website established by NRC for this rulemaking at http://ruleforum.llnl.gov.

The following sections set forth the resolution of the public comments.

A. General comments

Many commenters expressed strong support for the rule to improve the regulations in § 50.44 and “commend[ed] the NRC for developing a rule based on risk-informed and performance-based insights that would eliminate unnecessary regulatory requirements.” One industry commenter indicated that this rule will enhance public health and safety because it increases the reliability of the hydrogen and oxygen monitoring systems. The Advisory Committee on Reactor Safeguards (ACRS) stated that the draft proposed rulemaking for risk-informed revisions to 10 CFR 50.44 will provide more effective and efficient regulation to deal with combustible gases in containments.

The NRC also received feedback on several issues for which comments were specifically requested in the draft rule language. The existing rule provides detailed, prescriptive instructions using American Society of Mechanical Engineers (ASME) references for analyzing the performance of boiling water reactor (BWR) Mark III and pressurized water reactor (PWR) ice condenser containments. In the final rule, the NRC has provided an option for a more performance-based approach, which received positive public comment. Based upon stakeholder input, the final rule eliminates the existing references to ASME standards and other prescriptive requirements. The regulatory guide attached to this paper includes the ASME approach as one in which the intent of the regulations could be satisfied.

One private citizen questioned why the NRC was considering relaxing requirements that provide protection against some of the uncertainties and hazards of nuclear power. A citizen group opposed the changes by contending that eliminating the design-basis accident release, relaxing safety classifications, and relaxing licensee commitments to certain design and qualification criteria only benefits the money interests of the licensees. This group also stated its belief that the NRC's reliance on limited Three Mile Island (TMI) data points was insufficient to relax requirements solely to accommodate industry cost cutting strategies.

The NRC is moving to risk-informed, performance-based regulation that takes into account the benefits and consequences of actions by licensees and the NRC. One of the benefits of risk-informed regulation is that it concentrates resources on areas that are more important and minimizes resource allocation on areas that are shown to be less significant. As part of the basis for deciding the level of importance of various areas, during the 1980s and 1990s, the NRC sponsored a severe accident research program to improve the understanding of core melt phenomena, combustible gas generation, transport, and combustion, and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (e.g., NUREG-1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light water reactors. The result of these studies has been an improved understanding of combustible gas behavior during severe accidents and confirmation that the combustible gas release postulated from a design-basis LOCA was not risk-significant because it would not lead to early containment failure, and that the risk associated with gas combustion was from beyond-design-basis (e.g., severe) accidents.

In making its regulatory decisions, the NRC first considers public safety, then other issues such as public confidence and reducing unnecessary regulatory burden. Based upon the results of significant research into design-basis and beyond design-basis accidents, the NRC has determined that a design-basis combustible gas release is not risk-significant and certain beyond design-basis combustible gas releases are risk-significant. Therefore, the NRC is removing the requirements for combustible gas control systems that mitigate consequences of non-risk-significant design-basis accidents which are also not effective in reducing the risk from combustible gas releases in beyond-design-basis accidents.

The citizen group also contended that because GSI-191, "Assessment of Debris Accumulation on PWR Sump Pump Performance", is not resolved, removing the hydrogen recombiner requirements and relaxing the hydrogen and oxygen monitoring requirements are premature and constitute a dangerous trend towards risk “misinformed” regulation.

The NRC disagrees with the commenter’s contention. The NRC’s philosophy on all GSIs is to first determine whether the existing situation provides adequate protection of public health and safety, and if there is sufficient margin to allow continued safe operation of the affected plants while seeking a final resolution of the GSI. For GSI-191, the NRC concluded that even though uncertainties remained regarding the debris accumulation issue, adequate protection of public health and safety was maintained. Accordingly, the fact that GSI-191 has not reached final resolution does not present an impediment to the revision to § 50.44.

An industry group requested that the terms "safety-significant" and "industrial" instead of high and low safety/risk significance be used in this rule and regulatory guide. The NRC disagrees. The terms "high and low safety/risk significance" were not included in the proposed rule and are not in the final rule. The term "safety-significant", when used in supporting documentation, is used to identify systems, structures, and components (SSCs) that contribute to safety. The term does not confer the level of significance on the SSC. Additionally, the term "risk significant" is used to identify those conditions that contribute to risk. Again, no level of significance is assigned by the use of this term. Additionally, the change in terminology requested by the commenter would be inconsistent with the supporting NRC documents and reports. Changing terminology could cause unnecessary confusion on the part of licensees and the public.

B. General clarifications

One commenter questioned if the draft regulatory guide would become Regulatory Guide 1.7, Revision 3. When the NRC resolves the comments on DG-1117, the guidance will be published as Regulatory Guide 1.7, Revision 3.

A licensee requested that the first sentence of Item 3 of the fourth paragraph of section B of the draft regulatory guide be revised to read: "The following requirements apply to all construction permits or operating licenses under 10 CFR Part 50, and to all design approvals, design certifications, or combined licenses under 10 CFR Part 52, any of which are issued after the effective date of the rule." The NRC agrees that the commenter's request represents a clearer way of expressing the NRC's intent. In addition, the term "manufacturing licenses" has been added to make clear that the revised requirements apply to applicants for manufacturing licensees, which was inadvertently omitted from the proposed rule. These changes have been included in both the regulatory guide and in the final rule.

The licensee also requested that the NRC reword the statement in section 5 of the draft regulatory guide to read: “For future applicants and licensees as defined in Part 50.44(c), the analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.” Another licensee requested that section C.5, “Containment Integrity”, should state that it does not apply to currently licensed plants. The NRC disagrees with these requests. Section 5 of DG-1117 was intended to apply to current and future plants. However, the wording was not clear and inadvertently caused some confusion on the applicability of the section. To clarify that section 5 applies to current and future plants, its wording has been revised to more closely reflect the rule intent. This revision removes the following statements from the draft regulatory guide: “The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.” The above changes remove the misleading language and clarify the applicability of the section.

C. Monitoring systems

A private citizen expressed concern about the adequacy and survivability of non safety-related hydrogen and oxygen monitors for assessing hydrogen and oxygen levels after an accident. A reactor licensee stated that the changes to the requirements for hydrogen and oxygen monitoring would actually increase the reliability of hydrogen and oxygen monitoring equipment. A monitor vendor indicated that high-quality commercial grade hydrogen monitors may be susceptible to radiation-induced calibration degradation. The vendor also indicated that these monitors are susceptible to damage from aerosols released during the accident. The vendor believes that commercial grade detectors located inside containment would probably not function in a post-accident environment without verification testing and test-based modifications. The vendor claimed the more severe the accident, the less likely the sensors would properly operate due to increased radiation exposure and increased aerosol loading. In addition, the vendor believes that remote sampling lines for monitors located outside of containment are susceptible to clogging from high-solid aerosols. The vendor suggests it is prudent to retain the safety-related status of hydrogen monitors to ensure comprehensive qualification testing.

The NRC believes that the changes to the requirements for hydrogen and oxygen monitors will continue to ensure acceptable monitor performance. Should the changes result in a decrease in monitor reliability, it will not be significant and will not affect public health and safety because the functions served by the monitoring systems are not risk-significant for core melt accident sequences. This conclusion is supported by studies documented in the Feasibility Study (Attachment 2 to SECY-00-0198) which indicate the relatively low risk significance of monitoring systems. Because large, dry and sub-atmospheric containments are robust enough to withstand the effects of hydrogen combustion during full core melt accident sequences, hydrogen monitoring is not risk-significant for these containment designs. For BWR Mark I and Mark II containments, hydrogen monitoring systems are not risk-significant in the early stages of a core melt accident because these containments are inerted. For control of combustible gases generated by radiolysis in the late stage of a core melt accident, oxygen monitors are more important than hydrogen monitors for these designs. For this reason, the design and qualification requirements for oxygen monitors are more stringent than they are for hydrogen monitors. During core melt accidents in BWR Mark III and ice condenser containments, the hydrogen igniter systems are initiated by high containment pressure. Since hydrogen monitors are not needed to initiate or activate any mitigative features during these accidents, they are not risk-significant for reducing the combustible gas threat as long as the hydrogen igniters are operable. If the igniters are not operating (such as during station blackout) hydrogen monitoring does not reduce risk since the containment cannot be purged or vented without electrical power. Nevertheless, the amended rule requires licensees to retain hydrogen monitors (and oxygen monitors in Mark I and Mark II BWRs) for their containments because they are useful in implementing emergency planning and severe accident management mitigative actions for beyond design basis accidents.

As noted in sections III.C and D, as a consequence of eliminating the design-basis LOCA hydrogen release, the oxygen and hydrogen monitors are no longer required to mitigate potential consequences of combustible gases during design-basis LOCA accidents; thus the monitors are not required to be safety-related and need not meet the procurement, quality assurance, and environmental qualification requirements for safety-related components. Even though the final § 50.44 rule reclassifies requirements for monitoring systems, the hydrogen and oxygen monitoring systems are still required by the rule to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond-design-basis accident environment. Thus, licensees must consider the effects of radiation exposure and high-solid aerosols on monitor performance if they will be present in the post-accident environment for the specific type of facility and monitoring system design. The change made by the amended rule is that licensees are no longer required to use *only* safety-grade monitoring equipment. For a particular facility and monitoring system design, licensees will, in many cases, be able to select appropriate, high quality, commercial-grade monitors that will meet the performance requirements in the rule. In other cases, if no suitable commercial-grade monitors are available, safety-grade monitors may still be necessary. Also, since there are more types and designs of commercial-grade monitors available than there are safety-grade, the ability to use commercial-grade equipment may make it possible for licensees to select a better-suited monitor for their particular application. For example, it is stated in Attachment 2 to SECY-00-0198 that existing safety-grade hydrogen monitors have a limited hydrogen concentration range and are not the optimum choice. Commercial-grade monitors have the ability to monitor a wider range of hydrogen concentration and could be a better solution.

Since the amended rule implements a performance-based requirement for hydrogen and oxygen monitors to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond-design-basis accident environment, licensees will have to ensure that their procurement and quality assurance processes for such equipment address equipment reliability and operability in the beyond design basis accident environmental conditions for the specific facility and monitoring system design. Licensees who do not consider reliability and operability in appropriate environmental conditions when designing and procuring monitoring equipment, could be found by NRC inspectors to be in violation of the amended rule.

Another vendor asked if additional requirements beyond commercial grade will be imposed on the monitor's pressure retaining components because the analyzer loop forms part of the containment boundary. The monitor's pressure retaining components must meet current regulations concerning containment penetrations. This vendor also asked if their conclusion that grab samples cannot replace continuous monitoring is correct. The NRC has determined that grab samples cannot replace continuous monitoring. However, grab samples may be taken to verify hydrogen concentrations in the latter stages of the accident response.

A vendor asked if two trains of equipment would be an appropriate solution for ensuring analyzer availability. The NRC cannot respond to such a question without more information about the reliability of each individual train. Licensees are required to meet the requirements of the rule. Individual licensees may determine how they will meet the functionality, reliability, and capability requirements of the rule, using appropriate guidance such as the regulatory guide and subject to NRC review and inspection.

A licensee requested that section C.2.2 of the draft regulatory guide indicate that oxygen monitors are only required for plants that inerted containments. The NRC agrees with the commenter that oxygen monitors are only required for inerted containments, but disagrees with the suggested addition. The first sentence of section C.2.2 already states: "The proposed Section 50.44 would require that equipment be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control." The final version of the regulatory guide continues to indicate that oxygen monitoring is only necessary for facilities that have inerted containments. Thus, the NRC believes that the existing guidance is sufficient. This licensee also requested that another statement in section C.2.2 of the draft regulatory guide regarding existing oxygen monitoring commitments be clarified to show that these systems meet the intent of the rule. The NRC agrees with the need for clarification. The statement has been revised to read: “Existing oxygen monitoring systems approved by the NRC prior to the effective date of the rule are sufficient to meet this criterion.”

D. Purge

A licensee stated that the (model) safety evaluation (SE) should address the acceptability of eliminating containment purge as the design basis method for post-LOCA hydrogen control. The NRC disagrees. The NRC model SE only addresses requirements in the standard technical specifications or licensee technical specifications (TS). In this case, the NRC model SE is for the elimination of the requirements of hydrogen recombiners, and hydrogen and oxygen monitors from the TS. Because containment purging requirements are not in the standard technical specifications or licensees’ technical specifications, the NRC model SE does not make conclusions regarding the acceptability of eliminating containment purging as the design basis method for post-LOCA hydrogen control. However, the following statement from the Statements of Considerations was added to the model SE to address the comment: “...the NRC eliminated the hydrogen release associated with a design-basis LOCA from § 50.44 and the associated requirements that necessitated the need for the hydrogen recombiners and the backup hydrogen vent and purge systems.”

E. Station Blackout/Generic Safety Issue 189

The citizens group stated that the proposed §50.44 should require the deliberate ignition systems in Mark III and ice condenser containments to be available during station blackout. This comment pertains to resolution of GSI-189. The NRC disagrees with the commenter. The evaluation and resolution of GSI-189 is ongoing and proceeding independently of the rule as noted in Section II of this Supplementary Information.

F. Containment structural uncertainties

The citizens group argues that the NRC does not have an adequate non-destructive tool to eliminate concerns that containments were built with voids in their walls, that all steel reinforcement bar was improperly installed during construction to ensure uniform structural integrity of containment walls, and that the concrete used in containment walls is of sufficient quality that leaching of containment walls has not weakened the structure. The commenter states that without such non-destructive tools, it is unreasonable to reduce the defense-in-depth strategy with the proposed rule. The commenter provided no technical basis or information to support the assertion that containments were inadequately constructed. The commenter also asserts that the proposed rule creates an undue risk to the public health and safety to solely accommodate the financial interest of the regulated industry. Again, no technical basis was provided to support the assertion of increased risk.

The NRC disagrees with the commenter. The NRC relies on several layers of protection to prevent, detect, and repair defects discovered during construction of concrete containments, including voids, improperly installed reinforcement bar, and low quality concrete. These layers of protection include:

(1) The implementation by the licensee of their NRC-approved 10 CFR Part 50, Appendix B, Quality Assurance (QA) program and the licensee's Quality Control (QC) program;

(2) The requirements of 10 CFR 50.55(e) that holders of Construction Permits identify, evaluate, and report defects and failures to comply with NRC requirements associated with substantial safety hazards to the NRC in a timely manner, generally within 60 days; and

(3) The verification by NRC inspectors as defined by the NRC's construction inspection program contained in NRC Inspection Manual Chapter 2512 that the construction is in accordance with approved design documents, that the licensee is properly and effectively implementing their QA/QC program, that construction defects are reported to NRC as required by 10 CFR 50.55(e), and that appropriate corrective actions are taken by the licensee.

Whenever there is a doubt about the proper locations of reinforcing bars, or voids in a concrete containment structure, appropriate non destructive examination methods and conservative analysis are used by the licensees to demonstrate that the containment and its vital components are able to perform their intended functions.

In addition, the pre-operational performance of the Structural Integrity Test (SIT) provides an added assurance by physically demonstrating the overall structural capability of a concrete containment. Also, 10 CFR 50.65, the maintenance rule, requires licensees to monitor the performance or condition of certain structures to provide reasonable assurance that the structures are capable of fulfilling their intended function throughout the life of the plant. Licensees must also periodically inspect and test their containments in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, and Appendix J to 10 CFR Part 50. Finally, at plants that have renewed their licenses, aging management programs are in effect to monitor containment structures to ensure that aging does not significantly degrade their functional capability.

G. PRA/Accident Analysis

An individual submitted questions in three areas. First, the commenter asked why the 30-minute initiation time for initiating hydrogen monitoring was overly burdensome and suggested that the proposed 90-minute initiation time was arbitrary. The NRC disagrees with the commenter. The 30-minute initiation time was developed following the TMI-2 accident based on engineering judgement on the time within which the hydrogen monitors needed to be made functional. Putting this equipment into service within 30 minutes, as directed in NUREG-0737, was found by some utilities during severe accident training (e.g., on nuclear power plant simulators) to be unnecessarily distracting to operators, because it took them away from more important tasks that needed to be implemented in the near term while the monitoring did not need to be initiated for a longer period. The NRC has determined that performance-based functional requirements rather than prescriptive requirements achieve the desired goal of hydrogen monitor functionality while giving licensees an opportunity to better use operators' time during an accident. The noted 90 minutes come from the time licensees found was needed to get the monitors running in a manner that still met the goal of monitoring hydrogen levels and allowed sufficient time for other operator actions based on severe accident emergency operating procedures. Thus, the 90 minute time period was a result of changing to a performance-based approach and was not arbitrarily specified as the time within which the operators had to act.

The individual also stated that the proposed rule was reducing “defense in depth” and that if a utility cannot afford to operate and maintain its nuclear power reactors with the requisite caution and oversight, then the utility should not operate them at all. The NRC disagrees with the commenter’s assertion that the amended regulations do not provide adequate defense-in-depth. Defense-in-depth continues to be a prime consideration in NRC decision making. The NRC makes its decisions considering public safety first. Only after public safety is ensured are other issues such as public confidence and reduction of unnecessary burden considered. Defense-in-depth is an element of the NRC’s safety philosophy that employs successive measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. It provides redundancy as well as the philosophy of a multiple-barrier approach against fission product releases. Defense-in-depth does not mean that equipment installed in a nuclear power plant never should be removed. Adequate defense-in-depth may be achieved through multiple means or paths.

The commenter also questioned whether the NRC staff has adequate data to demonstrate that the amount of residual and radiolytically-generated combustible gases generated during a design-basis LOCA would not be risk-significant -- especially if the LOCA occurred in a plant with older fuel and SSCs than were present during the accident at Three Mile Island, Unit 2. The NRC disagrees with the commenter’s assertion that insufficient information is known about hydrogen generation to support amending the current regulations. The amount of hydrogen generated during a design-basis LOCA is not affected by the relative age or vintage of reactor fuel or SSCs. The NRC has developed significant data and insights on the behavior of design-basis and severe accidents after the TMI-2 accident. In amending § 50.44 in 1985, the NRC recognized that an improved understanding of the behavior of accidents involving severe core damage was needed. During the 1980s and 1990s, the NRC devoted significant resources and sponsored a severe accident research program to improve the understanding of core melt phenomena; combustible gas generation, transport, and combustion; and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (e.g., NUREG-1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light water reactors. The result of these studies has been an improved understanding of combustible gas behavior during severe accidents. One of the insights from these studies is confirmation that the hydrogen release postulated from a design-basis LOCA was not risk-significant because it would not lead to early containment failure. In addition, it was found that the vast majority of the risk associated with hydrogen combustion was from beyond design-basis (e.g., severe) accidents. The amended requirements are based on the NRC's careful consideration of the post-Three Mile Island information.

H. Passive autocatalytic recombiners

An individual questioned why the United States was allowing the removal of recombiners while the French are requiring the installation of passive autocatalytic recombiners in their reactors. The NRC has determined that passive autocatalytic recombiners (PARs) do not need to be considered for U.S. PWRs with large-dry containments or sub-atmospheric containments. This conclusion was drawn after applying the quantitative and qualitative criteria in the form of a framework for risk-informed changes to technical requirements of 10 CFR Part 50 (See attachment 1, SECY-00-0198). The NRC found that hydrogen combustion is not a significant threat to the integrity of large, dry containments or sub-atmospheric containments when compared to the 0.1 conditional large release probability of the framework document. In SECY-00-0198, the NRC also concluded that additional combustible gas control requirements for currently licensed large-dry and sub-atmospheric containments were unwarranted.

I. Reactor venting

An individual expressed concern for the elimination of the requirement prohibiting venting the reactor coolant system if it would aggravate the challenge to containment. According to the comment, the venting could cause an increase in the radiological effluents released off site and an increase in public exposure. The NRC disagrees with the individual's conclusion. As noted in section III.F of this Supplementary Information, the requirement to install high point vents was imposed by the 1981 amendment to § 50.44. This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.

For certain severe accident sequences, the use of reactor coolant system high point vents is intended to reduce the amount of core damage by providing an opportunity to restore reactor core cooling. Although the release of noncondensible and combustible gases from the reactor coolant system could, in the short term, "aggravate" the challenge to containment, the use of these vents will positively affect the overall course of the accident. The release of combustible gases from the reactor coolant system has been considered in the containment design and mitigative features that are required for combustible gas control. Any venting is highly unlikely to affect containment integrity or cause an increase in the radiological effluents released off site that could potentally increase public radiation exposure. However, such venting may reduce the likelihood of further core damage. The reduction in core damage would reduce both the generation of combustible gases and the magnitude of the radiological source term that could be released, thus reducing the potential for public exposure.

An industry organization requested a revision in a statement in section III.F in the statement of considerations (SOC) concerning the purposes of the high point vents from: "...venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged core and thus prevents further accident progression" to "...the purpose of the high point venting is to ensure that natural circulation cooling is an option for maintaining a long term safe stable state following a core damage accident in which significant amounts of noncondensible gases, such as hydrogen might be generated and retained in the reactor coolant system.” The NRC disagrees with the comment and believes the current wording is adequate. Other information in section III.F adequately defines the purpose of high point vents by acknowledging their usefulness *both* for forced circulation scenarios and in the natural circulation mode.

J. Design Basis Accident hydrogen source term

A private citizen questioned that because an unexpected hydrogen bubble and an unexpected hydrogen burn occured during the accident at Three Mile Island, should hydrogen buildup be considered a known risk for which licensees should try to monitor and control as thoroughly as possible? The NRC agrees with the commenter that hydrogen generation during severe accidents is an expected phenomenon. After the TMI accident, the NRC has sponsored an extensive research program on the behavior of severe accidents. This program was designed improve the understanding of core melt phenomena; combustible gas generation, transport, and combustion; and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (e.g., NUREG-1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for water-cooled reactors.

The result of these studies has been an improved understanding of combustible gas behavior during severe accidents and confirmation that the combustible gas release postulated from a design-basis LOCA was not risk-significant because it would not lead to early containment failure, and that the risk associated with gas combustion was from beyond-design-basis (e.g., severe) accidents. Thus, the requirements for control and monitoring of combustible gases are being reduced for the non-risk-significant design-basis accident scenarios. The amended regulations are entirely consistent with and justified by the findings of the post- TMI studies.

K. Requested minor modifications

An industry group requested that the last paragraph of Section B of the draft regulatory guide be changed to read: "The treatment requirements for the safety-significant components in the combustible gas control systems, the atmospheric mixing systems and the provisions for measuring and sampling are delineated in Section C, Regulatory Position." The NRC disagrees with the requested change. Section 50.44 is being revised to eliminate unnecessary requirements relating to combustible gas control in containment. The remaining requirements have been determined by the NRC to be necessary to mitigate the risk associated with combustible gas generation. The regulatory guide provides recommended treatments for all structures, systems, and components credited for meeting those requirements. Because the regulatory guide is only guidance, licensees are free to devise their own treatments for these structures, systems, and components, subject to NRC review and inspection.

L. Atmosphere mixing

A private citizen suggested adding criteria to the regulatory guide to assess the adequacy of the performance of atmosphere mixing systems. The NRC disagrees with the commenter that these criteria are needed. The NRC has already evaluated the adequacy of atmosphere mixing at currently operating pressurized and boiling water reactors. However, for future water-cooled reactor designs, the NRC has decided to specify that containments must have the capability for ensuring a mixed atmosphere during "design-basis and significant beyond design-basis accidents". Other guidance on determining the adequacy of atmosphere mixing systems is also provided in the rule and the regulatory guide.

An industry group requested that the SOC and regulatory guide be revised to only impose requirements on safety-significant hydrogen (atmospheric) mixing systems. They contend that some large dry containments have hydrogen mixing systems in addition to containment fan cooler units. The fan cooler units are supposedly the prime mode of ensuring a mixed atmosphere; therefore, the hydrogen mixing systems are classified as low safety-significance. The industry group believes that regulatory requirements should not be imposed on low safety-significant equipment. The NRC disagrees with the requested change. Section 50.44 is being revised to eliminate unnecessary requirements relating to combustible gas control in containment. The remaining requirements have been determined by the NRC to be necessary to mitigate the risk associated with combustible gas generation. The regulatory guide provides recommended treatments for all structures, systems, and components credited for meeting those requirements. Because the regulatory guide only provides guidance, licensees are free to devise their own treatments for these structures, systems, and components, subject to NRC review and inspection.

M. Current versus future reactor facilities

An industry group requested that § 50.44(c) be amended to clarify that its requirements relate only to light-water reactors. The NRC acknowledges that the proposed requirements in § 50.44(c) were largely patterned after light-water reactor requirements and might not be specifically applicable to all types of future light-water and non light-water reactor designs. Therefore, the NRC has modified § 50.44(c) to apply only to future water-cooled reactors with characteristics such that the potential for production of combustible gases during design-basis and significant beyond design-basis accidents is comparable to current light-water reactor designs. In addition, the NRC has added a new paragraph (d) that specifies combustible gas control information to be provided by applicants for future reactor designs when the potential for the production of combustible gases is not comparable to current light-water reactor designs. The purpose of this information is to determine if combustible gas generation is technically relevant to the proposed design; and, if so, to demonstrate that safety impacts of combustible gases generated during design-basis and significant beyond design-basis accidents have been addressed in the design of the facility to ensure adequate protection of public health and safety and common defense and security.

The industry group also commented that the regulatory guide is unclear on what parts are applicable to existing reactors and what parts are applicable to future reactors. The Introduction and section B do not agree. The NRC agrees. The regulatory guide has been modified to clarify the applicability of the revised § 50.44 to present and future water-cooled and non water-cooled reactors. The industry group also noted that the proposed language, the draft regulatory guide, and the proposed change to the Standard Review Plan incorrectly assume that all new reactor designs will be light-water reactors and will present the same combustible gas hazard. Future reactors, whether light-water or non-light-water may use different materials, cooling, or moderating mediums that may not result in the production of the same combustible gases, or quantities of combustible gas as the current light-water reactor designs. The NRC agrees. For the reasons given above, the final rule, the regulatory guide, and the standard review plan have all been modified to clarify their applicability to future reactor designs.

N. Equipment qualification/survivability

A licensee suggested adding a clarifying statement to the SOC concerning equipment survivability for Mark III and ice condenser plants. The commenter requested a statement clearly stating that no new equipment survivability requirements are being imposed and that existing equipment survivability and environmental analyses remain valid for compliance with the revised rule. The NRC agrees with commenter that the rule does not impose any additional equipment survivability requirements on licensees; existing equipment survivability and environmental analyses remain valid. The hydrogen and oxygen monitoring systems are required by the rule to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond design-basis accident environment.

This licensee also noted that, due to the reclassification of the hydrogen and oxygen monitors from RG 1.97 Category I to lower categories, these monitors no longer have to be qualified in accordance with 10 CFR 50.49. The NRC agrees that the monitoring equipment need not be qualified in accordance with § 50.49. The hydrogen and oxygen monitoring systems are still required by the rule to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond design-basis accident environment.

The licensee suggested that the NRC clarify that the revised rule will not affect the requirements or environmental conditions used by licensees to demonstrate compliance with § 50.49. The NRC agrees with the commenter that existing licensee analyses and environmental conditions used to establish compliance with 10 CFR 50.49 will not be affected by the amended rule and that no new analyses or environmental conditions are imposed by these amendments to § 50.44.

V. Petitions for Rulemaking - PRM-50-68

The NRC received a petition for rulemaking submitted by Bob Christie of Performance Technology, Knoxville, Tennessee, in the form of two letters dated October 7, 1999, and November 9, 1999. The petition requested that the NRC amend its regulations concerning hydrogen control systems at nuclear power plants. The petitioner believes that the current regulations on hydrogen control systems at some nuclear power plants are detrimental and present a health risk to the public. The petitioner believes that similar detrimental situations may apply to other systems as well (such as the requirement for a 10-second diesel start time). The petitioner believes his proposed amendments would eliminate those situations associated with hydrogen control systems that present adverse conditions at nuclear power plants. The petition was docketed as PRM-50-68 on November 15, 1999. On January 12, 2000 (65 FR 1829), the NRC published a notice of receipt of this petition in the *Federal Register* that summarized the issues it contains.

Specifically, the petitioner performed a detailed review of the San Onofre Task Zero Safety Evaluation Report (Pilot Program for Risk-Informed Performance-Based Regulation) conducted by the NRC staff and dated September 3, 1998, concerning that plant's hydrogen control system. The petitioner requested that the NRC:

1. Retain the existing requirement in §50.44(b)(2)(i) for inerting the atmosphere of existing Mark I and Mark II containments.

2. Retain the existing requirement in §50.44(b)(2)(ii) for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding.

3. Require all future light water reactors to postulate a 75 percent metal/water reaction (instead of the 100 percent required by the current rule) for analyses undertaken pursuant to §50.44(c).

4. Retain the existing requirements in §50.44 for high point vents.

5. Eliminate the existing requirement in §50.44(b)(2) for a mixed atmosphere in containment.

6. Eliminate the existing requirement for hydrogen releases during design basis accidents of an amount equal to that produced by a metal/water reaction of 5 percent of the cladding.

7. Eliminate the requirement for hydrogen recombiners or purge in LWR containments.

8. Eliminate the existing requirements for hydrogen and oxygen monitoring in LWR containments.

9. Revise GDC 41 -- Containment Atmosphere Cleanup -- to require systems to control fission products and other substances that may be released into the reactor containment for accidents only where there is a high probability that fission products will be released to the reactor containment.

10. Issue an interim policy statement applicable to all NRC staff to ensure that the NRC Executive Director for Operations was promptly notified whenever staff discovered cases where compliance with design-basis accident requirements was detrimental to public health.

The NRC received five comment letters on PRM-50-68. The commenters included two nuclear power plant licensees, a nuclear reactor vendor, a nuclear power plant owners group, and the Nuclear Energy Institute (NEI). Copies of the public comments on PRM-50-68 are available for review in the NRC Public Document Room, 11555 Rockville Pike, Rockville, Maryland. All commenters were supportive of some of the issues raised by the petition. One of the reactor licensees commented that analytical and risk bases exist to support the proposed changes for Mark I Boiling Water Reactor containments. The other licensee endorsed the comments submitted by NEI. The reactor vendor commented that the petitioner's proposal simplifies the language and requirements of the regulation while retaining an equivalent level of safety. However, the vendor also noted that the proposal does not appear to address the structural integrity of the containment as in the existing language at §50.44(c)(3)(iv). The owner's group commented that the changes requested by the petitioner for large, dry containments were also applicable to ice condenser containments and suggested that the requirement for all hydrogen control measures in §50.44 be reexamined and made "consistent with many other portions of plant operation and maintenance." The NEI agreed with the petitioner that the San Onofre hydrogen control licensing actions could be applied generically for pressurized water reactors with large, dry (including subatmospheric) containments. One licensee, the reactor vendor and the NEI disagreed with the petitioner's position that an interim policy statement is necessary to instruct the NRC staff how to proceed in instances when "adherence to design basis requirements would be detrimental to public health." The other commenters were silent regarding the request for an interim policy statement.

The NRC has evaluated the technical issues and the associated public comments and has determined that the specific issues contained in PRM-50-68 should be granted in part and denied in part as discussed in the following paragraphs.

Issue 1: Retain the existing requirement for inerting the atmosphere of existing Mark I and Mark II containments.

Resolution of Issue 1: Consistent with the petitioner's request, §50.44(b)(2)(i) of the final rule retains the current requirement for inerting of existing Mark I and Mark II containments. The NRC's basis for this decision is provided in section III A of this document.

Issue 2: Retain the existing requirement for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding.

Resolution of Issue 2: Consistent with the petitioner's request, § 50.44(b)(2)(ii) of the final rule retains the above requirement for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding. The NRC's basis for this decision is provided in section III A of this document.

Issue 3: Require all future light water reactors to postulate a 75 percent metal/water reaction (instead of the 100 percent required by the current rule) for analyses under § 50.44(c).

Resolution of Issue 3: The NRC declines to adopt this request. For future water-cooled reactors, the final rule retains the previous requirement to postulate hydrogen generation by a

100 percent metal/water reaction when performing structural analyses of reactor containments under accident conditions. Future containments that cannot structurally withstand the consequences of this amount of hydrogen must be inerted or must be equipped with equipment to reduce the concentration of hydrogen during and following an accident. The NRC's basis for this decision is provided in section III E of this document.

Issue 4: Retain the existing requirements for high point vents.

Resolution of Issue 4: Consistent with the petitioner's request, the requirements for high point vents in former 10 CFR 50.44(c)(3)(iii) have been retained in the final rule, but have been modified slightly to clarify the acceptable use of these vents during and following an accident. Because the need for high point vents is relevant to ECCS performance during severe accidents and is not pertinent to combustible gas control, these high point venting requirements have been removed from 10 CFR 50.44 and relocated to 10 CFR 50.46a where the remaining requirements for ECCS are located. The basis for this decision is provided in section III F. of this document.

Issue 5 Eliminate the existing requirement in § 50.44(b)(2) to ensure a mixed atmosphere in containment.

Resolution of Issue 5: The NRC declines to adopt this request. The final rule retains the requirement for all containments to ensure a mixed atmosphere to prevent local accumulation of combustible or detonable gasses that could threaten containment integrity or equipment operating in a local compartment. The NRC's basis for retaining this requirement is provided in section III A of this document.

Issue 6: Eliminate the existing requirement for postulating design basis accident hydrogen releases of an amount equal to that produced by a metal/water reaction of 5 percent of the cladding.

Resolution of Issue 6: The NRC grants this request. The NRC has determined that hydrogen release during design basis accidents is not risk-significant because it does not contribute to the conditional probability of a large release of radionuclides up to approximately 24 hours after the onset of core damage. The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by implementation of severe accident management guidelines. The NRC's technical basis for eliminating this requirement is discussed in greater detail in section III B of this document.

Issue 7: Eliminate the requirement for hydrogen recombiners or purge in light-water reactor containments.

Resolution of Issue 7: The NRC grants this request. As noted in Issue 6 above, the NRC has determined that hydrogen release during design basis accidents is not risk-significant because it does not contribute to the conditional probability of a large release of radionuclides up to approximately 24 hours after the onset of core damage. The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by implementation of severe accident management guidelines. Thus, hydrogen recombiners and hydrogen vent and purge systems are not required. The NRC's basis for eliminating these requirements is discussed in greater detail in section III B of this document.

Issue 8: Eliminate the existing requirements for hydrogen and oxygen monitoring in light-water reactor containments.

Resolution of Issue 8: The NRC declines to adopt this request. The final rule retains the existing requirement for monitoring hydrogen in the containment atmosphere for all plant designs. Hydrogen monitors are required to assess the degree of core damage during beyond design-basis accidents. Hydrogen monitors are also used in conjunction with oxygen monitors to guide licensees in implementation of severe accident management strategies. Also, the NRC has decided to codify the existing regulatory practice of monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. If an inerted containment became de-inerted during a beyond design-basis accident, other severe accident management strategies, such as purging and venting, would need to be considered. Monitoring of both hydrogen and oxygen is necessary to implement these strategies. The NRC's bases for these requirements are discussed in greater detail in sections III C. and III D of this document.

Issue 9: Revise GDC 41 -- Containment Atmosphere Cleanup -- to require systems to control fission products and other substances that may be released into the reactor containment for accidents only when there is a high probability that fission products will be released to the reactor containment.

Resolution of Issue 9: The NRC declines to adopt the petitioner's request on this issue. The NRC believes that the amended rule alleviates the need to revise Criterion 41. In a December 4, 2001, letter from the petitioner to the NRC, the petitioner inferred that the intent of the proposed change was to focus Criterion 41 on the containment capability when a severe accident occurs. This concern is addressed in the final § 50.44 that establishes the design criteria for reactor containment and associated equipment for controlling combustible gas released during a postulated severe accident. The General Design Criteria in Appendix A of 10 CFR Part 50 were established to set the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The postulated accidents used in the development of these minimum design criteria are normally design-basis accidents. The NRC believes it is not appropriate to address severe accident design requirements in the General Design Criteria.

Issue 10: The petitioner requested the NRC to issue an interim policy statement applicable to the NRC staff to ensure that the NRC Executive Director for Operations was promptly notified whenever the staff discovered cases where compliance with design-basis accident requirements was detrimental to public health.

Resolution of Issue 10: The petitioner's additional request for an interim policy statement is not part of the petition for rulemaking. Nevertheless, the NRC has evaluated the request and associated public comments and has concluded that hydrogen control requirements referenced by the petitioner have been modified in the final rule so that design basis requirements ensure adequate protection of public health and safety. The NRC also believes that if NRC staff members discover future situations when design basis requirements detract from safety, the staff will elevate these issues for management review; thus, no NRC staff guidance in this area is necessary.

Petition for Rulemaking - PRM-50-71

The NRC also received a petition for rulemaking submitted by NEI. The petition, dated April 12, 2000, was published in the *Federal Register* for public comment on May 31, 2000 (65

FR 34599). The petitioner requested that the NRC amend its regulations to allow nuclear power plant licensees to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have been approved by the NRC staff. The petitioner believes the proposed amendment would improve the efficiency of the regulatory process by eliminating the need for individual licensees to obtain exemptions to use advanced cladding materials that have already been approved by the NRC.

Specifically, the petitioner states that the NRC's current regulations require uranium oxide fuel pellets, used in commercial reactor fuel, to be contained in cladding material made of Zircaloy or ZIRLO. The petitioner indicates that the requirement to use either of these materials is stated in § 50.44 and § 50.46. The petitioner notes that subsequent to promulgation of these regulations, commercial nuclear fuel vendors have developed and continue to develop materials other than Zircaloy or ZIRLO that the NRC reviews and approves for use in commercial power reactor fuel. Each of these approvals requires the NRC to grant an exemption to the licensee that requests to use fuel with these cladding materials. The petitioner requests that the NRC amend its regulations to allow licensees discretion to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided that the cladding materials meet the fuel cladding performance requirements and have been reviewed and approved by the NRC staff. The petitioner notes that during the past nine years there have been at least eight requests for exemptions and that each exemption has cost more than $50,000. The petitioner states that the requests for exemptions have become increasingly more frequent, causing significant administrative confusion and having a potentially adverse effect on efficient and effective use of NRC, licensee, and vendor resources.

The petitioner believes the NRC should amend § 50.44 and § 50.46 to allow the use of other zirconium-based alloys in addition to those specified in the current regulations. The petitioner states that the stated goal of the existing regulations is to ensure adequate cooling for reactor fuel in case of a design-basis accident. However, the petitioner asserts that the proposed amendment does not degrade the ability to meet that goal. The petitioner believes it removes an unwarranted licensing burden without increasing risk to public health and safety.

The NRC received 11 comment letters on PRM 50-71. Seven comments were from nuclear reactor licensees, two from individual members of the public, one from a nuclear reactor vendor and one from a nuclear industry trade association (NEI). Five of the nuclear reactor licensees were supportive of the petition and endorsed the comments and positions provided by NEI in their comments on the petition. One licensee stated that the proposed rule should note that if a fuel vendor's cladding has met the requirements for use on a generic basis, a process for the implementing utility to use that fuel under their existing license already exists. Another licensee agreed that industry needs relief on use of zirconium-based cladding, but because cladding is a critical safety barrier, the basis for relief should come from proven, in-reactor performance. A better approach would be to update the approved list of allowed fuel rod cladding materials as more products demonstrate reliable, in-reactor performance.

Two comments were received from individuals. One individual opposed the petition because it did not contain the specific review and acceptance criteria that NRC would utilize when reviewing and approving future cladding materials under the proposed rule. The commenter also opposed the practice of allowing lead fuel assembly tests to demonstrate performance of new materials in commercial reactors before NRC approval, but also stated that long term performance testing of materials was necessary, must take into account any differences at individual utilities, and must consider future performance in dry cask storage systems. Another individual commented that the petition should be denied because the evaluations of cladding materials do not account for the realities of plant operation under normal conditions and the loss of coolant accident environment. This commenter stated that NRC approval of materials whose properties fell "within" acceptance criteria was unacceptable because an approval might be issued for a material whose properties were "right to the limit" without an adequate margin of safety. With respect to hydrogen generation, the commenter opposed generic approvals of new materials because site-specific material variations might yield unexpected results.

The nuclear reactor vendor supported adoption of the proposed rule changes published in the Federal Register and agreed with the suggested revision of § 50.46(e) proposed by NEI in its comments on the document. The vendor also recommended consideration of a direct final rule process to implement the petition. The NEI provided revised wording for proposed language in § 50.46(e) and urged the NRC to promulgate the revision as a direct final rule.

After evaluating the petition and public comments, the NRC has determined that the petition should be denied in part. The final § 50.44 rule has been written so that it does not refer to specific types of zirconium cladding; instead, the rule applies to all boiling and pressurized water reactors. When the NRC approves the use of boiling or pressurized water reactor fuel with other types of cladding, no exemptions from § 50.44 will be needed. Thus, even though the final rule does not contain the language specifically requested to be added by the petitioner, the rule accomplishes the petitioner's intended purpose with respect to § 50.44. Also, the NRC did not utilize the direct final rulemaking process because the other provisions being amended in § 50.44 were too complex to allow the promulgation of a direct final rule. The NRC is making no decision at this time on the part of the petition regarding the request to amend the regulations in § 50.44 to allow the use of other zirconium-based alloys in addition to those specified in the current regulations. The NRC will evaluate that portion of the NEI petition in a separate action.

VII. Section-by-Section Analysis of Substantive Changes

Section 50.34 - Contents of applications; technical information.

Paragraph (a)(4) on ECCS performance is revised to reference the reactor coolant system high point venting requirements located in § 50.46a. These requirements were relocated to § 50.46a from § 50.44.

Paragraph (g) is redesignated as paragraph (h) and a new paragraph (g) is added, that requires applications for future reactors to include the analyses and descriptions of the equipment and systems required by § 50.44.

Section 50.44 - Combustible gas control in containment.

Paragraph (a), *Definitions*. Paragraph (a) adds definitions for two previously undefined terms, "mixed atmosphere," and "inerted atmosphere."

Paragraph (b), *Requirements for currently-licensed reactors*. This paragraph sets forth the requirements for control of combustible gas in containment for currently-licensed reactors. All BWRs with Mark I and II type containments are required to have an inerted containment atmosphere, and all BWR Mark III type containments and PWRs with ice condenser type containments are required to include a capability for controlling combustible gas generated from a metal water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment integrity. Current requirements in § 50.44(c)(i), (iv), (v), and (vi) are incorporated in to the amended regulation without substantial change. Previously reviewed and installed combustible gas control mitigation features to meet the existing regulations are considered to be sufficient to meet this section. Because these requirements address beyond design-basis combustible gas control, it is acceptable for structures, systems, and components provided to meet these requirements to be non safety-related and may be procured as commercial grade items.

Paragraph (b)(1), *Mixed atmosphere*. The requirement for capability ensuring a mixed atmosphere in all containments is consistent with the current requirement in § 50.44(b)(2) and does not require further analysis or modifications by current licensees. The intent of this requirement is to maintain those plant design features (e.g., availability of active mixing systems or open compartments) that promote atmospheric mixing. The requirement may be met with active or passive systems. Active systems may include a fan, a fan cooler, or containment spray. Passive capability may be demonstrated by evaluating the containment for susceptibility to local hydrogen concentration. These evaluations have been conducted for currently licensed reactors as part of the IPE program.

Paragraph (b)(3) retains the existing requirements for BWR Mark III and PWR ice condenser facilities that do not use inerting to establish and maintain safe shutdown and containment structural integrity to use structures, systems, and components capable of performing their functions during and after exposure to hydrogen combustion.

Paragraph (b)(4)(i) codifies the existing regulatory practice of monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. The rule does not require further analysis or modifications by current licensees but certain design and qualification criteria are relaxed. The rule requires that equipment for monitoring oxygen be functional, reliable and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a beyond design-basis accident. Equipment for monitoring oxygen must perform in the environment anticipated in the severe accident management guidance. The oxygen monitors are expected to be of high-quality and may be procured as commercial grade items. Existing oxygen monitoring commitments for currently licensed plants are sufficient to meet this rule.

Paragraph (b)(4)(ii) retains the requirement in § 50.44(b)(1) for measuring the hydrogen concentration in the containment. The rule does not require further analysis or modifications by current licensees but certain design and qualification criteria are relaxed. The rule requires that equipment for monitoring hydrogen be functional, reliable and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident of comparable severity to the accident at Three Mile Island. Equipment for monitoring hydrogen must perform in the environment anticipated in the severe accident management guidance. The hydrogen monitors may be procured as commercial grade items. Existing hydrogen monitoring commitments for currently licensed plants are sufficient to meet this rule.

Paragraph (b)(5) retains the current analytical requirements in § 50.44(c)(3)(iv) that BWR Mark III and PWR ice condenser containments be provided with a hydrogen control system justified by a suitable program of experiment and analysis that can handle without loss of containment integrity an amount of hydrogen equivalent to that generated by a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel. Existing licensee hydrogen control systems and analyses are expected to be sufficient to demonstrate compliance with this requirement.

Paragraph (c), *Requirements for future water-cooled reactor applicants and licensees*. Paragraph (c) promulgates requirements for combustible gas control in containment for all future water-cooled reactor construction permits or operating licenses under Part 50 and for all water-cooled reactor design approvals, design certifications, combined licenses, or manufacturing licenses under Part 52, whose reactor designs have comparable potential for the production of combustible gases as current light water reactor designs. The current requirements in § 50.34(f)(2)(ix) and (f)(3)(v) are retained without material change, but have been consolidated and reworded to be more concise. Paragraph (c)(1) requires a mixed containment atmosphere during design-basis and significant beyond design-basis accidents. This wording was chosen to specify a mixed atmosphere requirement during important accident scenarios similar to the current requirements for PWR and BWR containments. Paragraph (c)(2) requires all containments to have an inerted atmosphere or limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad coolant reaction, uniformly distributed, to less than 10 percent and maintain containment structural integrity and appropriate accident mitigating features. Structures, systems, and components (SSCs) provided to meet this requirement must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Equipment survivability expectations under severe accident conditions should consider the circumstances of applicable initiating events (such as station blackout[1] or earthquakes) and the environment (including pressure, temperature, and radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of design-specific reviews which include probabilistic risk-assessment as required by § 52.47(a)(v). Because these requirements address beyond design-basis combustible gas control, SSCs provided to meet these requirements need not be subject to the environmental qualification requirements of § 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; and redundancy/diversity requirements of 10 CFR Part 50, Appendix A. Guidance such as that found

[1]Section 50.44 does not require the deliberate ignition systems used by BWRs with Mark III type containments and PWRs with ice condenser type containments to be available during station blackout events. The deliberate ignition systems should be available upon the restoration of power. Additional guidance concerning the availability of deliberate ignition systems during station blackout sequences is being developed as part of the NRC review of Generic Safety Issue 189: “Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident.”

in Appendices A and B of RG 1.155, "Station Blackout," is appropriate for equipment used to mitigate the consequences of severe accidents. Paragraph (c) also promulgates requirements for ensuring a mixed atmosphere and monitoring oxygen and hydrogen in containment, consistent with the requirements for current plants set forth in paragraphs (b)(1), and (b)(4)(i) and (ii).

Paragraph (d), *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees*. A new paragraph (d) is added to specify information that must be submitted by future reactor applicants to determine if combustible gas generation is technically relevant to the proposed design. If combustible gas generation is technically relevant, the applicant must submit additional information to demonstrate that safety impacts of combustible gases generated during design-basis and significant beyond-design-basis accidents have been addressed in the design of the facility to ensure adequate protection of public health and safety and common defense and security.

Paragraph (d) is applicable to non water-cooled reactors and water-cooled reactors that have different characteristics regarding the production of combustible gases from current light water reactors. The information must address the potential for producing combustible gases during design basis accidents and significant beyond design-basis accidents comparable to accident scenarios that were evaluated for combustible gas generation at current light water reactors.

Section 50.46a - Acceptance criteria for reactor coolant system venting systems.

Section 50.46a is a new section that contains the relocated requirements for high point vents currently contained in § 50.44. The amendment includes a change that eliminates a requirement prohibiting venting the reactor coolant system if it could “aggravate” the challenge to containment. Any venting is highly unlikely to affect containment integrity; however, such venting will reduce the likelihood of further core damage. The NRC continues to view use of the high point vents as an important strategy that should be considered in a plant’s severe accident management guidelines.

Section 52.47 - Contents of applications.

Section 52.47 is amended to eliminate the reference to paragraphs within § 50.34(f) for technically relevant requirements for combustible gas control in containment for future design certifications. Under the final rule, the technical requirements for combustible gas control will be set forth in § 50.44, rather than in § 50.34(f).

VIII. Availability of Documents

The NRC is making the documents identified below available to interested persons through one or more of the following methods as indicated.

Public Document Room (PDR). The NRC Public Document Room is located at One White Flint North, Public File Area O 1F21, 11555 Rockville Pike, Rockville, Maryland.

Rulemaking Website (Web). The NRC's interactive rulemaking Website is located at http://ruleforum.llnl.gov. These documents may be viewed and downloaded electronically via this Website.

NRC's Electronic Reading Room (ERR). The NRC's public electronic reading room is located at www.nrc.gov/NRC/ADAMS/index.html. (Provide accession number for each document.)

The NRC staff contact (NRC Staff). Richard Dudley, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone (301) 415-1116; e-mail rfd@nrc.gov.

Document	**PDR**	**Web**	**ERR**	**NRC Staff**
Comments received	X	X	X	
Regulatory Analysis	X	X	ML031640482	
RG 1.7, Rev. 3	X	X	ML031640498 X	
Rev. SRP, Section 6.2.5	X	X	ML031640518 X	

A free single copy of Regulatory Guide 1.7 may be obtained by writing to the Office of the Chief Information Officer, Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or E-mail: DISTRIBUTION@nrc.gov or Facsimile: (301) 415-2289.

Copies of NUREGS may be purchased from The Superintendent of Documents, U.S. Government Printing Office, Mail Stop SSOP, Washington, DC 20402-0001; Internet: bookstore.gpo.gov; (202) 512-1800. Copies are also available from the National Technical Information Service, Springfield, VA 22161-0002; www.ntis.gov; 1-800-533-6847 or, locally, (703) 605-6000. Some publications in the NUREG series are posted at NRC's technical document Website www.nrc.gov/NRC/NUREGS/indexnum.html.

IX. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless using such a standard is inconsistent with applicable law or is otherwise impractical. In this final rule, the NRC is using the following Government-unique standard: 10 CFR 50.44, U.S. Nuclear Regulatory Commission, October 27, 1978 (43 FR 50163), as amended. No voluntary consensus standard has been identified that could be used instead of the Government-unique standard.

X. Finding of No Significant Environmental Impact: Environmental Assessment

The NRC has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. The basis for this determination reads as follows:

This action endorses existing requirements and establishes regulations that reduce regulatory burdens for current and future licensees and consolidates combustible gas control regulations for future reactor applicants and licensees. This action stems from the NRC's ongoing effort to risk-inform its regulations. The final rule reduces the regulatory burdens on present and future power reactor licensees by eliminating the LOCA design-basis accident as a combustible gas control concern. This change eliminates the requirements for hydrogen recombiners and hydrogen purge systems and relaxes the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their safety and risk significance.

This action does not significantly increase the probability or consequences of an accident. No changes are being made in the types or quantities of radiological effluents that may be released off site, and there is no significant increase in public radiation exposure because there is no change to facility operations that could create a new or affect a previously analyzed accident or release path. There may be a reduction of occupational radiation exposure since personnel will no longer be required to maintain or operate, if necessary, the hydrogen recombiner systems which are located in or near radiologically controlled areas.

With regard to non-radiological impacts, no changes are being made to non-radiological plant effluents and there are no changes in activities that would adversely affect the environment. Therefore, there are no significant non-radiological impacts associated with the proposed action.

The primary alternative to this action would be the no action alternative. The no action alternative would continue to impose unwarranted regulatory burdens for which there would be little or no safety, risk, or environmental benefit.

The determination of this environmental assessment is that there is no significant offsite impact to the public from this action.

The NRC requested the views of the States on the environmental assessment for this rule. No comments were received.

XI. Paperwork Reduction Act Statement

This final rule decreases the burden on new applicants to complete the hydrogen control analysis required to be submitted in a license application, as required by sections 50.34 or 52.47. The public burden reduction for this information collection is estimated to average 720 hours per request. Because the burden for this information collection is insignificant, Office of Management and Budget (OMB) clearance is not required. Existing requirements were approved by the Office of Management and Budget, approval numbers 3150-0011 and 3150-0151.

XII. Public Protection Notification

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

XIII. Regulatory Analysis

The NRC has prepared a regulatory analysis on this regulation. The analysis examines the costs and benefits of the alternatives considered by the NRC. The regulatory analysis is available as indicated under the Availability of Documents heading of the Supplementary Information section.

XIV. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

XV. Backfit Analysis

The NRC has determined that the backfit rule does not apply to this final rule; and therefore, a backfit analysis is not required for this final rule because these amendments do not impose more stringent safety requirements on 10 CFR Part 50 licensees. For current licensees, the amendments either maintain without substantive change existing requirements or provide voluntary relaxations to current regulatory requirements. Voluntary relaxations (i.e., relaxations that are not mandatory) are not considered backfitting as defined in 10 CFR 50.109(a)(1). For future applicants and future licensees, the amendments also do not involve backfitting as defined in 10 CFR 50.109(a)(1) because the changes have only a prospective effect on future design approval and design certification applicants and future applicants for licensees under 10 CFR Part 50 and 52. As the Commission has indicated in other rulemakings, sec., e.g., 54 FR 15372, April 18, 1989 (Final Part 52 Rule), the expectations of future applicants are not protected by the Backfit Rule. Therefore, the NRC has not prepared a backfit analysis for this final rule.

XVI. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and record keeping requirements.

10 CFR Part 52

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and record keeping requirements, Standard design, Standard design certification.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR Parts 50 and 52.

PART 50 -- DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951, as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 - 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. In § 50.34, paragraph (a)(4) is revised, paragraph (g) is redesignated as paragraph (h), and a new paragraph (g) is added to read as follows:

§ 50.34 Contents of applications; technical information.

(a) * * *

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.

* * * * *

(g) *Combustible gas control*. All applicants for a reactor construction permit or operating license under this part, and all applicants for a reactor design approval, design certification, or license under part 52 of this chapter, whose application was submitted after [EFFECTIVE DATE

OF RULE], shall include the analyses, and the descriptions of the equipment and systems required by § 50.44 as a part of their application.

*   *   *   *   *

3. Section 50.44 is revised to read as follows:

<u>§ 50.44 Combustible gas control for nuclear power reactors.</u>

(a) *Definitions*.

(1) *Inerted atmosphere* means a containment atmosphere with less than 4 percent oxygen by volume.

(2) *Mixed atmosphere* means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

(b) *Requirements for currently-licensed reactors*. Each boiling or pressurized water nuclear power reactor with an operating license on [EFFECTIVE DATE], except for those facilities for which the certifications required under §50.82(a)(1) have been submitted, must comply with the following requirements, as applicable:

(1) *Mixed atmosphere*. All containments must have a capability for ensuring a mixed atmosphere.

(2) *Combustible gas control*.

(i) All boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere.

(ii) All boiling water reactors with Mark III type containments and all pressurized water reactors with ice condenser containments must have the capability for controlling combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity.

(3) *Equipment Survivability.* All boiling water reactors with Mark III containments and all pressurized water reactors with ice condenser containments that do not rely upon an inerted atmosphere inside containment to control combustible gases must be able to establish and

maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

(4) *Monitoring*.

(i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

(ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

(5) *Analyses*. Each holder of an operating license for a boiling water reactor with a Mark III type of containment or for a pressurized water reactor with an ice condenser type of containment, shall perform an analysis that:

(i) Provides an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75 percent of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include consideration of hydrogen control measures as appropriate;

(ii) Includes the period of recovery from the degraded condition;

(iii) Uses accident scenarios that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core.

(iv) Supports the design of the hydrogen control system selected to meet the requirements of this section; and,

(v) Demonstrates, for those reactors that do not rely upon an inerted atmosphere to comply with paragraph (b)(2)(ii) of this section, that:

(A) Containment structural integrity is maintained. Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff in accordance with § 50.90. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on; and

(B) Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including local detonations, unless such detonations can be shown unlikely to occur.

(c) *Requirements for future water-cooled reactor applicants and licensees*.[1] The requirements in this paragraph apply to all water-cooled reactor construction permits or operating licenses under this part, and to all water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 of this chapter, any of which are issued after [EFFECTIVE DATE].

(1) *Mixed atmosphere*. All containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents.

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[1]The requirements of this paragraph apply only to water-cooled reactor designs with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of [EFFECTIVE DATE OF RULE].

(2) *Combustible gas control*. All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

(3) *Equipment Survivability*. Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.

(4) *Monitoring*.

(i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

(ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

(5) *Structural analysis*. An applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant

reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

(d) *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees.* The requirements in this paragraph apply to all construction permits and operating licenses under this part, and to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 of this chapter, for non water-cooled reactors and water-cooled reactors that do not fall within the description in paragraph (c), footnote 1 of this section, any of which are issued after [EFFECTIVE DATE OF RULE]. Applications subject to this paragraph must include:

(1) Information addressing whether accidents involving combustible gases are technically relevant for their design, and

(2) If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public health and safety and common defense and security.

4. Section 50.46a is added to read as follows:

§ 50.46a Acceptance criteria for reactor coolant system venting systems.

Each nuclear power reactor must be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. High point vents are not required for the tubes in U-tube steam generators. Acceptable venting systems must meet the following criteria:

(a) The high point vents must be remotely operated from the control room.

(b) The design of the vents and associated controls, instruments and power sources must conform to Appendix A and Appendix B of this part.

(c) The vent system must be designed to ensure that:

(1) The vents will perform their safety functions, and

(2) There would not be inadvertent or irreversible actuation of a vent.

PART 52-EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER PLANTS

5. The authority citation for Part 52 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat.936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846).

6. In § 52.47, paragraph (a)(1)(ii) is revised to read as follows:

§ 52.47 Contents of applications

(a) * * *

(1) * * *

(ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) except paragraphs (f)(1)(xii), (f)(2)(ix) and (f)(3)(v);

* * * * *

Dated at Rockville, Maryland, this __th__ day of ____, 2003.

For the Nuclear Regulatory Commission.

____
Annette Vietti-Cook,
Secretary of the Commission.

ATTACHMENT 4: FINAL REGULATORY ANALYSIS

Regulatory Analysis for 50.44

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Tables

1. Statement of the Problem and Objective

Since the 1987 revision of 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors,” there have been significant advances in our understanding of the risk from nuclear power plants, in particular risk arising from the production and combustion of hydrogen (and other combustible gases) during reactor accidents. These advances are described in SECY-00-0198, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)” [1]. This new understanding has led to a reconsideration of the bases for the requirements in 10 CFR 50.44. A portion of this reconsideration is the proposed “rebaselining” of 50.44, as described in SECY-01-0162 [2]. This risk-informed, performance-based rulemaking is the subject of the regulatory analysis.

The objective of this regulatory analysis is to address the regulatory relaxation issues associated with the proposed rebaselining action described in [2], consistent with the regulatory analysis guidance documents [3, 4].

Two options are presented in [2]:

Option 1

Update the existing rule and delete the hydrogen recombiner requirements for all containment types. As a part of this rulemaking, additional changes to the regulations may be necessary to retain hydrogen monitoring requirements for accident assessment purposes. In addition, complete the resolution of GI-189.

Option 2

Update the existing rule and delete the hydrogen recombiner requirements for all facilities except those with BWR Mark III and PWR ice condenser containments. As a part of this rulemaking, additional changes to the regulations may be necessary to retain hydrogen monitoring requirements for accident assessment purposes. In addition, for the BWR Mark III and PWR ice condenser facilities, defer any rule changes until the staff completes its resolution of GI-189.

(Note that Generic Issue 189 (GI-189) will assess the costs and benefits of possible additional hydrogen control requirements for PWR ice condenser and BWR Mark III containment designs. Analyses indicate these containments are more susceptible to failure during station blackout sequences where the AC powered igniters are not available. Therefore, removing the dependence on AC power for the combustible gas control systems could be of value for risk-significant accidents.)

The first option was recommended because it presents the most complete, expeditious, and efficient method of updating the regulations, and therefore will be the subject of this regulatory analysis. As such, the regulatory analysis will focus on the recommended removal of hydrogen recombiner requirements and the relaxation of hydrogen monitoring requirements, as well as the relaxation of oxygen monitoring requirements for BWRs with Mark I and Mark II containments.

The issue of resolution of GI-189 is separate from the “rebaselining” of 50.44 and will be considered under a separate regulatory analysis.

Regarding the recombiners and their associated vent/purge systems, the staff has applied the risk-informed process described in Attachment 2 [5] to SECY-00-0198 [1] to each of the generic containment design types. The staff found that the outcome for PWR large dry and subatmospheric containment designs and for BWR Mark I and II containment designs was always the same. That is, for these containment types, the outcomes were that hydrogen recombiners could be eliminated from the design basis and no additional hydrogen control requirements would be needed. The outcome of the SECY-00-0198 process is less clear for PWR ice condenser and BWR Mark III containment designs. With respect to the need for recombiners however, the outcome was similar to that for other containment designs. That is, recombiners could be eliminated from the design basis of facilities with these containment designs with no significant risk impact. Other issues associated with the control of combustible gases for core melt accidents for these containment types are being deferred to the GI-189. A remaining issue for Mark I and Mark II type plants with inerted containments, is the potential for the production of oxygen by radiolysis during severe accidents to form combustible mixtures with hydrogen that has evolved from radiolysis and zirconium/water reactions. Although analysis indicates that it will take days for these combustible mixtures to develop, there is a concern with removing recombiners that could prevent combustion events that lead to containment failure. This concern is addressed in the regulatory analysis.

Regarding hydrogen monitoring, the analyses from SECY-00-0198 [1, 5] further concluded that hydrogen monitors at some facilities are not necessary for combustible gas control. However, these monitors, depending on plant type, may be needed to support emergency operating procedures, severe accident guidelines, and accident assessment functions that facilitate emergency response decision making. If these monitors are determined to be necessary only for accident assessment purposes, then this equipment would no longer be required to be safety grade. Therefore, updating hydrogen monitoring requirements could result in a reduction in unnecessary burdens in the areas of procurement, upgrading, and maintenance of hydrogen monitoring systems by reclassifying the monitors from an indication that most directly indicates the accomplishment of a safety function to backup and diagnostic instrumentation. Guidance on design specifications is delineated in Regulatory Guide 1.97 [6]. The guide specifies that safety-grade (Category 1) instrumentation provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power.

1.1 Background of Problem

1.1.1 History

In a June 8, 1999 Staff Requirements Memorandum (SRM) on SECY-98-300, “Options for Risk-Informed Revisions to 10 CFR Part 50,” the Commission approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50.

The staff provided its plan and schedule for the study phase of its work to risk-inform the technical requirements of 10 CFR Part 50 in SECY-99-264, “Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50,” dated November 8, 1999. The plan consists of two phases: an initial study phase (Phase 1), in which an evaluation of the feasibility of risk-informed changes along with recommendations to the Commission on proposed changes will be made; and an implementation phase (Phase 2), in which changes recommended from Phase 1, and approved by the Commission, will be made. SECY-99-264 discussed Phase 1 of the plan. In Phase 1, the staff is studying the ensemble of technical requirements contained in 10 CFR Part 50 to (1) identify candidate changes to requirements and design basis accidents (DBAs), (2) prioritize candidate changes to requirements and DBAs, and (3) establish the feasibility of and identify recommended changes to requirements.

The Commission approved proceeding with the proposed staff plan in an SRM dated February 3, 2000. In addition, the Commission directed the staff to highlight any policy issues for Commission resolution as early as possible during the process, particularly those related to the concept of defense-in-depth. Staff has been directed to develop a communication plan that facilitates greater stakeholder involvement and actively seeks stakeholder participation.

Revision of combustible gas control requirements following a postulated LOCA was requested in conjunction with Task Zero of the Risk-Informed, Performance-Based Regulation Pilot Program. This program was an initiative undertaken by the NRC and the Nuclear Energy Institute to improve the incorporation of risk-informed and performance-based insights into the regulation of nuclear power plants. Task Zero resulted in an exemption from combustible gas control requirements from the San Onofre nuclear generating station's design basis as documented in a letter to the licensee, dated September 3, 1999.

On April 12, 2000, the staff provided its first status report on Phase 1 in SECY-00-0086 ("Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3)") and also indicated its intention to expedite recommendations for risk-informed changes to 10 CFR 50.44 ("Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors").

On September 14, 2000, the staff provided its second status report on Phase 1 in SECY-00-0198 [1]. This SECY included a "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50" as Attachment 1 [7] and "Feasibility Study for a Risk-Informed Alternative to 10 CFR 50.44" as Attachment 2 [5]. In SECY-00-0198, the staff proposed a risk-informed voluntary alternative to the current 10 CFR 50.44. Attachment 2 [5] to that SECY described a process by which licensees could determine which of a number of possible regulatory requirements would apply to their facility, if they chose the voluntary alternative.

Since it completed SECY-00-0198, the staff has taken three actions that affect its approach and schedule for risk informing 10 CFR 50.44. First, the staff has continued the technical work described in the paper to develop hydrogen source terms and to assess the significance of seismically-initiated and fire-initiated accidents. Second, it established Generic Issue 189 (GI-189) to assess the costs and benefits of possible additional hydrogen control requirements for PWR ice condenser and BWR Mark III containment designs. (The issue raised in SECY-00-0198 was that analyses indicate these containments have a high conditional containment failure probability associated with station blackout sequences during which the AC powered igniters are not available. Therefore, removing the dependence on AC power for the combustible gas control systems could be of value for risk-significant accidents.) Third, the staff has applied the process described in Attachment 2 to SECY-00-0198 to each of the generic containment design types and concluded that hydrogen recombiners could be eliminated from the design basis for all LWRs and no additional hydrogen control requirements would be needed for any LWR, except those with ice condenser or MARK-III containments. SECY-01-0162 recommended removing this issue of additional hydrogen control measures for plants with ice condenser or Mark III containments from the rulemaking and assigning it to GI-189. With the removal of this issue from the rulemaking, the staff concluded that, for all containment types, a more efficient regulatory approach than that proposed in SECY-00-0198 would be to modify (rebaseline) the current 50.44 to eliminate the requirement for recombiners rather than offering a voluntary alternative that would, upon licensee evaluation, lead to the same result. Adopting this simplified approach could also help expedite the schedule for this rulemaking.

The analyses from SECY-00-0198 further concluded that hydrogen monitors are not risk-significant for combustible gas control. However, these monitors, depending on plant type, are needed to support emergency operating procedures, severe accident guidelines, and accident assessment functions that facilitate emergency response decision making. If these monitors are determined to be necessary only for these purposes, then this equipment would no longer be required to be safety grade. Therefore, unnecessary burden reduction benefits of updating hydrogen monitoring requirements could be realized in the areas of procurement, upgrading, and maintenance of these systems.

SECY-01-0162 [2] requests Commission approval of the staff's plans for proceeding with rulemaking to risk-inform 10 CFR 50.44, as requested in the SRM to SECY-00-0198, dated January 19, 2001. The SRM directed the staff to proceed expeditiously with rulemaking on the risk-informed alternative to 10 CFR 50.44, including completing outstanding technical work and necessary regulatory analyses. The Commission requested that the staff avoid overly prescriptive requirements and develop sufficiently flexible requirements to permit improvements in the methodology if better models become available. The Commission also directed the staff to provide recommendations for actions that could shorten the time for developing the proposed rule.

From these staff assessments, it was decided to proceed with the rebaselining of 10 CFR 50.44 with Option 1, described in Section 1, being the recommended option.

1.1.2 Contributions of Existing Requirements to the Problem

Recombiners are required to accommodate the amount of hydrogen associated with design basis events. Risk studies have shown that the risk is from beyond design basis events, not from the design basis events postulated in 10 CFR 50.44. For beyond design basis events, recombiners have little to no effect on mitigating the consequences of these events. The requirements for maintaining recombiners and hydrogen monitors as design-basis structures, systems and components (SSCs) have been burdensome to the nuclear power industry. Both the BWR Owners Group report [8] and Mr. R. Christie's Petition for Rulemaking [9] attest to this burden. This regulatory analysis takes into full account this burden in the Value-Impact portion of the analysis.

1.1.3 Immediate Problem as Part of Larger Issue and Ongoing Programs

This proposed regulatory action is the attempt to apply the staff's framework for risk-informing 10 CFR 50 and performance-basing any regulatory enhancements that might result. Next anticipated steps are to resolve GI-189 and to attempt to risk-inform 10 CFR 50.46.

1.1.4 Relationship of the Objectives of this Rulemaking to the Commission's Safety Goals

Since this action is a relaxation of requirements, it is neither a backfit nor subject to the safety goal requirements [3, page 9] normally imposed on regulatory actions. However, a level of assessment should be provided that demonstrates that the public health and safety and the common defense and security would continue to be adequately protected if the proposed reduction in requirements were implemented [3, page 6]. This demonstration is provided as part of Section 3 of this regulatory analysis. The risk analysis (described further in [5]) shows that these rulemaking actions either do not increase risk or only increase risk slightly, such that there is virtually no change in the conditions for assuring that the public health and safety is adequately protected.

In addition, a level of assessment should be provided that demonstrates that the cost savings attributed to the action would be substantial enough to justify taking action [3, page 6]. The assessment in Section 3 provides this demonstration.

1.1.5 Relationship to Formal Positions Adopted by National and International Standards Organization or Foreign Regulators

In a letter dated June 28, 2001, the French Nuclear Installations Safety Directorate directed Electricite de France to install passive autocatalytic recombiners (PARs) for severe accident hydrogen control in all PWR reactors by the end of 2007. This approach requires approximately 40 PARs per plant to achieve a capacity appropriate for severe accidents.

PARs will not be considered for US PWRs with large-dry containments or sub-atmospheric containments. This conclusion was drawn after applying the framework for risk-informed changes to the technical requirements of 10 CFR 50 [7]. The staff concluded that hydrogen combustion is not a significant threat to the integrity of these containment types, when compared to the 0.1 conditional large release probability of the framework document [7]. The staff further concluded that additional combustible gas control requirements for currently licensed large-dry and sub-atmospheric containments were unwarranted.

Based on available information, the staff concludes that the different position adopted by the French regulatory authority on severe accident hydrogen control stems from fundamental differences in their analysis and criteria for hydrogen sources and allowable buildup, treatment of random ignition of leaner mixtures, and different acceptance criteria for containment performance.

1.2 Backfit Rule

Since this regulatory analysis addresses only voluntary relaxations to the current rule, no backfit evaluation is required. Voluntary relaxations (i.e., relaxations that are not mandatory) do not fall within the definition of backfitting as defined in 10 CFR 50.109 (a)(1). As mandated on page 6 of NUREG/BR-0058, Revision 3, requirements associated with relaxations will be addressed, as described in Section 1.1.4 and in Section 3 of this regulatory analysis.

2. Identification and Preliminary Analysis of Alternative Approaches

The alternative approaches considered here are all based on variants of Option 1 of SECY-01-162, namely,

Update the existing rule and delete the hydrogen recombiner requirements for all containment types. As a part of this rulemaking, additional changes to the regulations may be necessary to retain hydrogen monitoring requirements for accident assessment purposes.

2.1 Approach 1: Option 1 of SECY-01-0162, With Relaxation for Hydrogen and Oxygen Monitoring

This approach will eliminate the requirement for recombiners and associated vent/purge systems for all containment types and will relax the requirements for hydrogen (& oxygen) monitoring from meeting Category 1 requirements, as defined in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," to meeting Category 3 for hydrogen, and Category 2 for oxygen.

The current special treatment requirements associated with the hydrogen and oxygen monitors are overly burdensome. Special treatment requirements associated with the hydrogen and oxygen monitors have been invoked by either order or commitments to NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.F.1, Attachment 6 which endorses RG 1.97 or RG 1.97 itself [6]. RG 1.97 recommends that the hydrogen and oxygen monitors be Category 1, which includes environmental qualification, seismic qualification, redundancy, being energized from station standby power sources, and being backed up by batteries where momentary interruption is not tolerable. Category 1 provides the most stringent requirements and is intended for key variables that most directly indicate the accomplishment of a safety function for design basis accident events. As discussed in SECY-00-198 [1], combustible gas control is not needed for a design-basis LOCA. Therefore, the hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. RG 1.97 states that Category 3 is intended to provide requirements that will ensure that high-quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. Hydrogen monitors can be backup instrumentation to support operator actions in the emergency operating procedures. Hydrogen monitors are used as diagnostic instrumentation to assess the degree of core damage, support severe accident guidelines, emergency operating procedures, and accident assessment functions that facilitate emergency response decision making. Therefore, Category 3 is a more appropriate categorization for hydrogen monitors.

The oxygen monitors also no longer meet the definition of Category 1 in RG 1.97. As discussed in SECY-00-198 [1, 5], oxygen monitoring is not needed to control combustible gas resulting from a LOCA. RG 1.97 states that Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status. Category 2 is a more appropriate categorization for the oxygen monitors because the oxygen monitors are used to indicate the status of the inerted containment environment, support severe accident guidelines, emergency operating procedures, and accident assessment functions that facilitate emergency response decision making.

Regarding recombiners, this rulemaking action will eliminate the requirement for combustible gas control systems following a postulated LOCA from §50.44 by the following means:

- Remove §50.44(c)(1) and §50.44(c)(2) — requires plants to demonstrate no uncontrolled hydrogen combustion following postulated LOCA but before operation of control system
- Remove §50.44(c)(3)(ii) including §50.44(c)(3)(ii)(A) and §50.44(c)(3)(ii)(B) — requires internal or external recombiners and imposes requirements on external recombiner containment penetrations
- Remove §50.44(d)(1) and §50.44(d)(2) — specifies the post-LOCA hydrogen amounts evolved in the accident.
- Remove §50.44(e), §50.44(f) and §50.44(g) — impose requirements relative to recombiners and purge-repressurization systems as means of hydrogen control following postulated LOCA
- Remove §50.44(h) — as all of the definitions it contains refer to text in earlier portions of the regulation that are already proposed to be deleted.

Some key implications of Approach 1 for NRC are summarized in Table 2.1 while some implications for industry are listed in Table 2.2. The tables present a screening assessment. Implications for both the industry and the NRC are evaluated in Section 3 in detail.

Table 2.1 Approach 1, NRC Implications

Item	Yes/No	Description/Comments
Rule change	Yes	10 CFR 50.44, will be revised by making the changes summarized above to the current requirements.
Impact on other regulations	Yes	NUREG-0737 and 10 CFR 50.34, 50.46a, and 52.47 will be revised to allow commercial grade monitors and to make related changes.
Revise/modify implementing documents	Yes	Existing regulatory guidance on safety grade monitors in Regulatory Guide 1.97 will be revised. Regulatory guidance on recombiners will need elimination.
Create implementing documents	Yes	New regulatory guides will be needed on providing acceptable methods for compliance with the risk-informed rule.
Analysis	No	No new analysis will be needed.
Review	Yes	Licensee submittals on hydrogen monitoring will need to be reviewed to verify compliance. License amendment requests associated with tech spec removal will be needed.
Inspection	Maybe	Depends on way in which compliance is achieved.

Table 2.2 Approach 1, Licensee Implications

Item	Yes/No	Description/Comments
Equipment	Maybe	Relaxation of special treatment requirements will allow for commercial grade monitors (Category 3 for hydrogen and Category 2 for oxygen). Changes will allow removal of recombiners, and purge/vent systems.
Analysis	No	No new analysis will be needed.
Maintenance/Inspection	Maybe	Will depend on the way compliance is achieved.
Tech Specs	Maybe	Remove hydrogen and oxygen monitors, recombiners, and vent/purge systems from technical specifications.
Procedures/Training	Maybe	Will depend on the way compliance is achieved.

2.2 Approach 2: Eliminate Requirement for Both Recombiners and Hydrogen Monitors

This second approach would then read as:

"Update the existing rule and delete the hydrogen recombiner requirements and hydrogen monitoring requirements for all containment types."

Under this approach, additional burden would be removed from the licensee by not having to install and maintain a (Category 3) hydrogen monitoring capability. However, then the hydrogen monitoring function would be lost for emergency planning and accident assessment functions.

2.3 Approach 3: Option 1 of SECY-01-0162, but Recombiner Requirements for BWRs with Mark I and Mark II Would Remain in Force

This third approach would then read as:

"Update the existing rule and delete the hydrogen recombiner requirements for all containment types, except Mark Is and Mark IIs. As a part of this rulemaking, additional changes to the regulations may be necessary to retain hydrogen monitoring requirements for accident assessment purposes."

Under this approach, continued burden (relative to Approach 1) would be required of licensees with plants that have Mark I or Mark II containments in that they would have to retain their recombiner capability. However, this approach would provide some control over the potential for very late containment failure that would otherwise result from combustion of gases produced from radiolysis following a severe accident (a de-inerting of the containment due to oxygen produced from radiolysis of water; a de-inerting that could be prevented by recombiners).

A variation on Approach 3 is to relax the current requirements for recombiners for plants with Mark I and Mark II containments, but still retain the recombiner function. Thus, for these plants, recombiners would be required, but they would no longer be safety-grade systems. The system design, operation and maintenance specifications would be relaxed, but would be sufficiently robust to meet reliability and availability guidelines. The values and the impacts associated with this variation on Approach 3 are intermediate between Approach 3–retain current recombiner requirements for plants with Mark I and Mark II containments, and Approach 1–remove recombiner requirements for plants with Mark I and Mark II containments. The “value” that this variation would provide is some control over the potential for very late containment failure by preventing late, large containment hydrogen burn events due to radiolysis, but with a cost (or impact) commensurate with maintaining the recombiner function. This is discussed in more detail in Section 4.

2.4 Approach 4: Base Reference Approach – No Change to Current Requirements

This approach allows for a baseline from which other approaches have been compared.

2.5 Discussion of Approaches

All of these approaches are variations on regulatory relaxations. All must pass the adequacy test which requires that the public health and safety and the common defense and security must continue to be adequately protected if the proposed reduction in requirements are implemented [3, page 6]. Approach 1 has been extensively evaluated, as summarized in [5].

Retaining Recombiners for Inerted Containments (Approach 3)

For the first 24 hours following initiation of core damage, the recombiners are ineffective -- either there is so much hydrogen present in containment that the recombiners are incapable of accommodating the hydrogen or the containment atmosphere is inert. The only question is whether there would be some use for the recombiners for containments in the long term recovery from an accident. Inerted containments could become de-inerted due to radiolysis under severe accident conditions occurring over a few days. PWR containments could use recombiners to remove residual hydrogen in the long term to prevent further hydrogen combustion. Consideration of these issues did not reveal any risk-significance. It is expected that accumulations of combustible gases beyond 24 hours can be managed by licensee implementation of SAMGs or other ad hoc actions because of the time available to take such action. This question is considered further in Section 3 of this regulatory analysis.

Comment on Retaining Purge/Vent or Venting Capabilities

In November 1994, the US nuclear industry committed to implement severe accident management at their plants by December 31, 1998, using the guidance contained in NEI 91-04, Revision 1, “Severe Accident Issue Closure Guidelines.” Generic severe accident management guidelines developed by each nuclear steam system supplier owners group include either purging and venting (for BWRs) or venting (for PWRs) the containment to address combustible gas control. The Commission continues to view purging and/or controlled venting of the containment to be important severe accident management strategies. This regulatory analysis does not evaluate such capabilities but assumes that licensees address purging and/or controlled venting of all containment types as a part of their severe accident management guidelines.

Approach 1 in this regulatory analysis concludes that the cost of maintaining the recombiners greatly exceeds the benefit of retaining them to prevent containment failure in sequences that progress to beyond 24 hours. The issue of eliminating the requirement for safety-grade purge/vent systems is not specifically analyzed in this regulatory analysis because the staff believes that the above conclusion would also be true for the backup hydrogen purge system. The cost is expected to exceed the estimated benefit of $21,320 as calculated in Appendix A of this document. In addition, the benefit would not be as great because the hydrogen purge system does not prevent a release. The hydrogen purge system would allow for a controlled release without containment failure as opposed to an uncontrolled release due to containment failure.

Eliminating Hydrogen Monitoring (Approach 2)

Combustible gas generation and combustion from beyond design basis accidents involving both fuel-cladding oxidation and core-concrete interaction has not been shown to be risk-significant when using the framework document's quantitative guidelines. The risk of early containment failure from hydrogen combustion is limited by the following mitigative features: (1) inerting in Mark I and II containments, (2) igniters in Mark III and ice condenser containments, and (3) the large volumes and likelihood of random ignition in large dry and sub-atmospheric containments that help prevent the build-up of detonable concentrations. Hydrogen monitoring is not needed to initiate or activate any of these measures, hence hydrogen monitors have a limited significance in mitigating the threat to containment in the early stages of a core melt accident.

Hydrogen monitors are needed to assess the degree of core damage and confirm that random or deliberate ignition has taken place and that containment integrity is not threatened by an explosive mixture. If an explosive mixture that could threaten containment integrity exists during a beyond design basis accident, then other severe accident management strategies, such as purging and/or venting, would need to be considered. For Mark I, II and III containments, the monitoring of hydrogen is used extensively in the emergency procedure guidelines/severe accident guidelines. On these bases, the Commission will require hydrogen monitoring for beyond design basis severe accident management in all containment designs. Hydrogen monitoring will be evaluated as part of this regulatory analysis. However, the staff notes that there have been arguments made that hydrogen monitors are not needed for these emergency planning purposes [9].

Both the industry and the NRC staff have determined the need for hydrogen monitoring for Severe Accident Management Guidelines (SAMGs) and emergency planning. For example, NEI 99-01, "Methodology for Development of Emergency Action Levels," recommends declaring a General Emergency when a radiation monitor reading corresponding to 20 percent fuel clad damage is registered. This corresponds to a hydrogen concentration inside containment of approximately 3-4 percent. The NRC Response Technical Manual, RTM-96, which is used for incident response, indicates that the concentration of containment hydrogen is more accurate than the containment radiation monitors whose ability to predict the degree of core damage is affected by fission product decay, shielding, and spray actuation. The GE, Westinghouse, and CE core damage assessment methodologies all include hydrogen monitoring. Hydrogen monitors are needed to confirm that random ignition has taken place and that containment venting does not need to be considered. Currently, severe accident management guidance includes consideration of venting based on containment pressure, hydrogen concentration, and radiation. This is a greater concern for Mark I and II plants that rely more heavily on hydrogen and oxygen monitoring to support actions such as RCS depressurization, spray initiation, and containment venting. Thus, removal of hydrogen monitoring will compromise emergency planning and severe accident management. Therefore, Approach 2 is screened out as an option and is not considered further in this regulatory analysis.

By retaining the requirement for hydrogen monitoring capability while at the same time relaxing the special treatment requirements, Approach 1 allows for more effective emergency planning capability and severe accident management, but also provides relief from regulatory burden.

2.6 Summary of the Preliminary Analysis of Alternative Approaches

Three approaches have been considered with reference to a no action baseline (Approach 4). The proposed rule as described in SECY-01-0162 is Approach 1. Approach 2 allows for removal of all hydrogen monitor requirements, not just a relaxation of requirements from Safety Grade (Category 1- Special Treatment) to Category 3. Approach 3 is the same as Approach 1 except it would not allow for the removal of recombiners for plants with Mark I or Mark II containments. There is a sufficient argument to screen out Approach 2, based on the utility of hydrogen monitoring for accident assessment functions that facilitate emergency response decision making and severe accident management, as supported by both the NRC and the industry. Relaxing the requirements for hydrogen monitoring should not compromise the utility of this monitoring capability as part of SAMGs. The subject of the following Value-Impact assessment then will be an analysis of Approaches 1 and 3, relative to taking no action (Approach 4).

3. Value-Impact Assessment

This section provides an assessment of the Values and the Impacts of the approaches discussed in Section 2, following the Regulatory Analysis guidance in [3, 4]. The two key issues, namely hydrogen monitoring and recombiners, are addressed separately. In Section 3.1, a summary of the Value-Impact assessment is provided. This is followed in Section 3.2 with comments on the assessment methodology and the assumptions used in the analysis. The required statement regarding the Safety Goal comprises Section 3.3. In Section 3.4, the Value-Impact analysis is presented.

3.1 Summary of Value-Impact Assessment

Section 3.4 provides an assessment of the values and impacts of the approaches discussed in Section 2. In Section 4, the results are presented. Tables 3.1, 3.2 and 3.3 summarize these results.

Table 3.1 Summary of the Value-Impact Assessment for Hydrogen Monitor Relaxation: Approach 1 compared to Baseline (Approach 4)

	per plant (average)	for Industry: 103 plants
Value	approximately zero	approximately zero
Impact	-$517,000	-$53,000,000
Value-Impact	$517,000	$53,000,000

Table 3.2 Summary of the Value-Impact Assessment for Recombiner Removal: Approach 1 compared to Baseline (Approach 4) for All PWRs and Mark III BWRs

	per plant (average)	for Industry: 69 PWRs, 4 BWRs
Value	$12,000	$876,000
Impact	-$438,000	-$31,974,000
Value-Impact	$450,000	$32,850,000

Table 3.3 Summary of the Value-Impact Assessment for Recombiner Removal: Approach 1 compared to Baseline (Approach 4) for Mark I and Mark II BWRs

	per plant (average)	for Industry (30 BWRs)
Value	$400	$12,000
Impact	-$437,500	-$13,125,000
Value-Impact	$438,000	$13,137,000

For both the monitors and the recombiners, the Value-Impact results are positive, indicating that this rulemaking action is supported by the Value-Impact assessment. Consideration of uncertainties in the assessment and consideration of the impact of Approach 3 – allowing recombiner removal only for PWRs and the BWRs with Mark III containments – does not alter the conclusion that the rulemaking action is justified. These matters are considered further in Sections 3.4 and 4.

3.2 Introduction to Value-Impact Assessment

This Value-Impact assessment follows the guidelines in [3, 4]. Consistent with these guidelines, the following assumptions are made in the assessment:

- The year chosen as a basis is 2002 and all costs are adjusted to reflect 2002 dollars
- The discount rate used is 7 percent, as recommended in [4]
- The remaining life of the average plant is assumed to be 35 years. This value was determined by adding 20 years (assuming license renewal) to 15 years remaining on the plant's current license [4]
- Using the 7 percent discount rate and 35-year lifetime, the multiplier used for determining the 2002 cost equivalent for yearly costs over the remaining life of the plant is 13.053 [4].

The "Values" considered in the quantitative assessment are:

- Public Health – Accident
- Public Health – Routine
- Occupational Health – Accident
- Occupational Health – Routine
- Property – Offsite
- Property – Onsite

The "Impacts" considered in the quantitative assessment are:

- Industry Implementation
- Industry Operation
- NRC Implementation
- NRC Operation

The sign convention, consistent with [4], is that increased public and occupational health (e.g., decreased risk to the public) and increased property values are "positive," while reduced public and occupational health (e.g., increased risk to the public) and reduced property values are "negative." Likewise, increased implementation and operation costs for the industry and NRC are "positive" while reduced implementation and operation costs (e.g., reductions in regulatory burden) for the industry and NRC are "negative."

The equation for determining the Value-Impact is then:

Value-Impact = {sum of all Values} - {sum of all Impacts} =

{(Public Health_Accident) + (Public Health_Routine) + (Occupational Health_Accident) + (Occupational Health_Routine) + (Property_Offsite) + (Property_Onsite)} – {(Industry Implementation) + (Industry Operation) + (NRC Implementation) + (NRC Operation)}

Thus, a positive Value-Impact will support a rulemaking action while a negative Value-Impact will not, independent of whether the rulemaking action is a relaxation or an enhancement.

3.3 Safety Goal Evaluation

As stated in Section 1.1.4, relaxations of requirements are not subject to the safety goal evaluation requirements.

3.4 Estimation and Evaluation of Values and Impacts for the Selected Alternatives

The Value-Impact assessment comprises two parts: 1) consideration of hydrogen monitoring, and 2) consideration of recombiners.

3.4.1 Hydrogen Monitoring

Regulatory actions that reduce current requirements (remove special treatment requirements) must be based on the determination that two conditions are satisfied:

- The public health and safety and the common defense and security would continue to be adequately protected if the proposed reduction in requirements or positions were implemented.
- The cost savings attributed to the action would be substantial enough to justify taking the action.

It has been determined that hydrogen monitoring is not needed to actuate the primary means for combustible gas control. Rather, its utility is for support of alternative EOP actions, emergency planning, and emergency decision making. The intent of the present approach is to allocate some performance to hydrogen monitoring as part of accident management. Accordingly, this regulatory analysis has already screened out Approach 2, which completely eliminates monitoring.

Approach 1 will relax the current requirements on hydrogen monitoring. The special treatment requirements on hydrogen monitoring currently in force can be relaxed if there is assurance that commercial-grade monitors can adequately meet the above-stated needs, and thereby provide assurance that the public health and safety and the common defense and security will continue to be protected. The high-level guidelines for performance-based regulatory activities show how to assess whether commercial-grade monitors can meet the present needs. Based on the low challenge frequency of this function (the frequency at which the hydrogen monitoring function is expected to be challenged), periodic verification of the functional capability of the hydrogen monitoring system is adequate, provided that the verification protocol tests the appropriate range of atmospheric conditions and that licensee corrective action programs include addressing issues in hydrogen monitoring performance if such issues arise. These detailed aspects are addressed in the regulatory guidance.

The cost savings per plant for this relaxation are estimated by the BWR Owners' Group [8] to be in the range of $40K to $150K per year for monitor maintenance, testing, and calibration costs. If these costs represent typical costs across the industry, yearly industry savings would range from $4M to $15M per year. If monitoring systems are replaced, the additional savings would be $400K to $900K per monitoring system replacement. However, there will be costs (impacts) associated with implementation of this rule change, as listed in Tables 2.1 and 2.2. All these costs (impacts) and cost savings (negative impacts) are described in more detail in Section 3.4.1.1 below.

3.4.1.1 Identification of Attributes

In the determination of the values and impacts of this proposed action, it should be noted that since this is a proposed relaxation, most attributes as defined in [4] will normally be "negative," since the risk will actually increase (most times only slightly) for items 1 through 4, and the impacts (items 7 through 10) will normally be negative (although there will be "positive" impact elements). The remaining attributes are presented qualitatively in Section 3.4.1.1.11. These attributes will be summarized and compared in Section 4. Below is a discussion of the Value-Impact attributes for hydrogen monitoring relaxation.

3.4.1.1.1 Public Health (Accident)

Consideration of the possible increase (or possible decrease) in risk to the public from relaxing the requirements for hydrogen monitoring is not subject to quantitative analysis. One aspect, however, can be discussed from a qualitative point of view.

By going from Category 1 requirements to Category 3, the monitors will not be subject to the Category 1 quality assurance requirements, redundancy requirements, Class 1E requirements or seismic requirements. Thus, for the purposes intended, namely, to assess the degree of core damage and confirm that random or deliberate ignition has taken place and that containment integrity is not threatened by an explosive mixture, the monitors might not be as reliable or available. This could complicate emergency decision making. In general, less information or misleading information would be expected to incur costs to the public in the form of the consequences of false-positive or false-negative evacuation decisions. Actual quantification of the value of degraded information depends on the details of procedures and guidelines, and the availability of alternative sources of information to support evacuation decisions, in addition to depending on the low frequency at which this information is needed. Any actual difference in the availability of the hydrogen monitoring function caused by a change in special treatment requirements would be difficult to establish, and its impact, most probably, would be negligible. Although not as stringent as Category 1, Category 3 is intended to ensure that high-quality off-the-shelf instrumentation is obtained and provides for servicing, testing and calibration.

3.4.1.1.2 Public Health (Routine)

There is no change in the Public Health (Routine), when comparing Approach 1 to the base case (Approach 4) since this approach does not involve any change to normal operational (routine) releases from the plant.

3.4.1.1.3 Occupational Health (Accident)

There is no change in the Occupational Health, when comparing Approach 1 to the base case (Approach 4) since the onsite damage from the accident and the resultant health effects would have occurred in any event.

3.4.1.1.4 Occupational Health (Routine)

This attribute is a value which accounts for radiological exposures to workers during normal facility operations. The proposed change seeks to relax the requirements for hydrogen and oxygen monitors. Typically, the hydrogen and oxygen monitors are located outside containment. Based on this, there would be very little change, if any, in the routine occupational health of the workers. In the event that a plant may have monitors located inside containment, the savings associated with no longer being required to perform certain surveillance would be minimal, but contribute to the overall benefits of the proposed change.

3.4.1.1.5 Offsite Property

As with consideration of risk to the public, consideration of the possible increase (or possible decrease) in offsite property costs resulting from relaxing the requirements for hydrogen monitoring is not subject to quantitative analysis. However, from a qualitative point of view, the arguments here for offsite property would be similar to those discussed in Section 3.4.1.1.1. Studies [10] have shown that the dollar equivalents for offsite property and public health (public risk impact) are the same order of magnitude. Thus, since the impact on public health is small, the impact on offsite property will also be small.

3.4.1.1.6 Onsite Property

There is no change in the Onsite Costs, when comparing Approach 1 to the base case (Approach 4) since the onsite damage from the accident would have occurred in any event.

3.4.1.1.7 Industry Implementation

This attribute is an impact which accounts for the projected net economic effect on the affected licensees to install or implement mandated changes. Approach 1 would relax the requirements for the hydrogen and oxygen monitors. As part of the relaxation, a new regulatory guide would be developed, or Regulatory Guides 1.7 and 1.97 would be revised, no longer requiring the monitors to be safety grade. Effectively, licensees could replace their Category 1 systems with Category 3 systems for hydrogen monitors and Category 2 systems for oxygen monitors. Although licensees would be able to meet the revised guidance with their current systems, it is likely that most licensees will replace their current monitors with more modern commercial grade models. Replacement costs would include modification package development, commercial grade monitors, removal and installation, and disposal. For recent severe accident mitigation alternative analysis, one PWR estimated [11] the cost to develop and implement an integrated hardware modification package, including post-implementation costs such as training, to be $70,000. The cost of commercial grade hydrogen monitors is estimated to be between $3,000 and $5,000 per sensing location. Using an example of 10 locations, this cost averages to be $40,000 per plant. Since the monitors are located outside containment, it is not certain whether any radioactive waste would be generated from the replacement of the monitors. Therefore, it is assumed to be small and costs for disposal are not estimated for this analysis.

Because the existing systems would satisfy the proposed regulation, it is expected that licensees would perform the modification during a regularly scheduled outage. Additionally, the monitoring systems are located outside containment (for most plants), so licensees could replace the systems while the plant is on-line, thus not necessitating an outage. At an estimated cost of $500K to $1M per day each day a plant is not operating, it is unlikely that any plant would extend an outage to perform this modification. Therefore, costs associated with shutdown and replacement power are not included.

The relaxation in Approach 1 will most likely precipitate a technical specification change. It will be to licensees' advantage to amend their technical specifications; therefore, licensees may incur a cost for preparing and submitting a license amendment request. According to NUREG/CR-4627 [12], it costs approximately $28,000 (adjusted to 2002 dollars) to prepare a typical uncomplicated technical specification amendment request. Since it is likely that licensees will submit one license amendment request that will cover both the monitors and the recombiners, only half of the cost ($14,000) for the amendment is considered in this portion of the Value-Impact analysis. See Section 3.4.2.2.7 for inclusion of the remaining half of this cost.

3.4.1.1.8 Industry Operation

This attribute is an impact which measures the projected net economic effect due to routine and recurring activities required by the proposed action on all affected licensees. According to industry estimates [8], it costs between $80,000 and $150,000 per year per reactor to operate and maintain hydrogen/oxygen monitors. Although this estimate is for a BWR, it is expected that costs for PWRs are similar. A relaxation of the requirement as recommended in Approach 1 is expected to reduce such costs by approximately 50 percent [8]. Assuming an annual cost of $100,000, a typical plant could realize savings of $50,000 per year, or $650,000 over the remaining life assumed by this analysis.

3.4.1.1.9 NRC Implementation

Approach 1 will necessitate a rulemaking as well as revision to or development of regulatory guidance. Whether or not the Commission chooses to proceed with the rulemaking, the costs associated with the development of the rulemaking and associated guidance are sunk costs, and not considered by this regulatory analysis.

Approach 1 involves the relaxation of a requirement which will result in the subsequent deletion of associated technical specifications. Therefore, license amendments are expected on the part of the licensees, i.e., licensees will request an amendment to delete requirements associated with operation and surveillance of the monitors. Therefore, the NRC will incur costs associated with review and approval of the amendment requests. According to NUREG/CR-4627 [12], it costs approximately $17,000 (adjusted to 2002 dollars) to review a typical uncomplicated technical specification amendment request. This cost includes preparation of a generic communication and model technical specification change. However, it should be noted that the technical specification amendment request for monitors is likely to be combined with the amendment request for the recombiners. Therefore, $8,500 is assumed for the hydrogen monitor portion of the Value-Impact.

3.4.1.1.10 NRC Operation

This attribute is an impact which measures the projected net economic effect on the NRC after the proposed action is implemented. As a result of the proposed action, there will be a reduced effort during inspections. This reduction is expected to be small, and not quantified for the purposes of this analysis.

3.4.1.1.11 Other Attribute Considerations

For completeness, the remaining attributes that make up the full set [4] are addressed here. Several – Safeguards and Security, Antitrust, Environmental, General Public, Improvement in Knowledge, and Other Government – have no bearing on this regulatory analysis and therefore are not discussed further. A discussion follows for the issue of Regulatory Efficiency.

One of the major motivations for this rulemaking is to reduce unnecessary regulatory burden on both the industry and the NRC. This is reflected in the preceding sections in reductions in the impacts, primarily for industry operations.

With relatively small industry and NRC implementation costs, savings to the industry in "Operation" drives the equation and allows for the conclusion that the benefits of the relaxation far outweigh the costs envisioned. Safety is not compromised because the monitors will be available when needed for severe accident management, with a functionality commensurate with the consequences and probability of severe accident events. Defense in depth is assured through other means of managing these accidents.

3.4.2 Recombiner Removal

This section focuses on the issue of removal of recombiners and associated vent/purge systems. The staff analysis, as presented in Attachment 2 to SECY-00-0198 [5], demonstrates that recombiners serve little or no safety function in plants with large dry, ice-condenser, or Mark III containments. They may have utility for plants with Mark I or Mark II containments a number of days after a severe accident as a means to accommodate oxygen generated by radiolysis. Approach 3 addresses the values and impacts of retaining recombiners for these plants. Table 3.4 summarizes the staff position.

Table 3.4   Staff Position on Means of Hydrogen Control

Containment Type	Means of Hydrogen Control	Comments
Large-Dry	No active means	Volume/strength sufficient to accommodate hydrogen threat
Ice Condenser	Hydrogen Igniters	Ignitors sufficient to accommodate hydrogen threat, except during station blackout–deferred to GI-189
Mark III	Hydrogen Igniters	Igniters sufficient to accommodate hydrogen threat, except during station blackout–deferred to GI-189
Mark I	Inerted Containment	Inerted containment sufficient to accommodate hydrogen threat, except possibly for long-term radiolysis
Mark II	Inerted Containment	Inerted containment sufficient to accommodate hydrogen threat, except possibly for long-term radiolysis

As noted in Section 3.4.1, regulatory actions that reduce current requirements must be based on the determination that two conditions are satisfied:

1. The public health and safety and the common defense and security would continue to be adequately protected if the proposed reduction in requirements or positions were implemented.
2. The cost savings attributed to the action would be substantial enough to justify taking the action.

The following value-impact assessment addresses both of these requirements. The assessment focuses on Approach 1. By separating out the assessment into two parts – (1) all PWR containments and all BWR Mark III containments and (2) all BWR Mark I & II containments, the value and impacts for Approach 3 can be more easily compared. This is because Approaches 1 and 3 are the same for all PWR containments and all BWR Mark III containments.

For Approach 1, the only increase in risk will come from not being able to accommodate combustible mixtures of oxygen and hydrogen in the long term for the Mark I and Mark II containments, if the recombiners were removed. In order to determine the magnitude of this risk increase, a baseline analysis was performed, as described in Section 3.4.2.1. This is followed by an assessment of the Value-Impact attributes that make up the Value-Impact determination, as described in Section 3.4.2.2.

3.4.2.1 Baseline Risk for the Mark I and Mark II Plants

Methodology

For the Mark I and Mark II analysis, Peach Bottom was selected as a representative plant. Relevant data on sequence frequencies and characterization, containment failure probabilities, radiological source terms to the environment, and risk consequences were obtained for Peach Bottom from a number of sources that were readily available and deemed best suited to the task, including plant-specific IPEs, IPEEEs, and a number of NUREG studies. For this plant type, the main challenge is posed by long-term generation of hydrogen and oxygen through radiolysis, and therefore risk-significant sequences are made up of all sequences that progress to the very late phase without containment failure or bypass.

A baseline risk was estimated for the risk-significant sequences using the available data, under the assumption that combustible gas control is unavailable for these sequences. Using the same sources of data, sensitivity case risk estimates were calculated assuming that some means of combustible gas control is available and 100 percent effective. These two calculations were the basis for obtaining a maximum achievable risk-benefit from their difference. Note that these calculations treat only the increased risk from offsite dose; offsite economic costs are addressed separately in Section 3.4.2.2.5. For a more detailed presentation of the methodology and data employed in performing these calculations, see Appendix A (BWR Mark I).

Results

Results of the risk-benefit calculations are described in detail in Appendix A. A summary of these results is shown below in Table 3.5. For BWRs with Mark I containments, the maximum risk-benefit from controlling the possible threat posed by radiolysis is estimated at $21,300. This figure includes both internal and external events (the latter made up mainly of fires).

Table 3.5 Summary of Risk-Benefit Results for Combustible Gas Control

Result	**BWR Mark I (Peach Bottom)**
CDF for Risk-Significant Events (events/reactor-year)	7.26e-6
Offsite Health Risk (whole-body person-rem per year within 50 miles)	
Baseline (without provision for combustible gas control)	0.82
Sensitivity (with provision for combustible gas control)	<0.001
Difference	0.82
Risk-Benefit ($)	
Baseline (without provision for combustible gas control)	$21,300
Sensitivity (with provision for combustible gas control)	very small
Difference	-$21,300[1]

1. Includes both internal and external events.

3.4.2.2 Identification of Attributes

Below is a discussion of the Value-Impact attributes for recombiner relaxation (considering both Approaches 1 and 3). These attributes will be summarized and compared in Section 4.

3.4.2.2.1 Public Health (Accident)

The decrease in public health due to this relaxation results in a numerical value of -$21,300 per plant for Approach 1 for BWRs with Mark I and Mark II containments, as described in Section 3.4.2.1. The value was determined by using the methodology described in Section 5.7.1 of [4]. It is the product of the person-rem/year (0.82), the monetary value of public health ($2,000/person rem), and the multiplier for present worth (13.05). This multiplier was calculated assuming a 7 percent discount rate and an average plant remaining lifetime of 35 years (starting in 2002). This lifetime was determined by subtracting 9 years from the 1993 data presented in Table B.1 of [4] -- remaining lifetime of 24 years -- and adding 20 years to account for license renewal.

There have been arguments posed by [9] that this “relaxation’ will improve safety. Basically the argument is that mandated hydrogen control activities (e.g., putting recombiners into operation during an accident and then monitoring them) could distract operators from more important tasks in the early phases of accident mitigation and could have a negative impact on the higher priority critical operator actions. The staff agrees that removal of recombiner requirements could have this safety benefit [13]. This benefit can not be quantified but should be considered in the uncertainty associated with -$21,300/plant.

Since Approach 3 does not alter the recombiner requirements for BWRs with Mark I and Mark II containments, the numerical value for decrease in public health is zero.

3.4.2.2.2 Public Health (Routine)

There is no change in the Public Health (Routine), when comparing Approach 1 or Approach 3 to the base case (Approach 4) since neither of these approaches involve any changes to normal operational (routine) releases from the plant.

3.4.2.2.3 Occupational Health (Accident)

There is no change in the Occupational Health, when comparing Approach 1 to the base case (Approach 4) since the onsite damage from the accident and the resultant health effects would have occurred in any event. This is also the case for Approach 3.

3.2.2.2.4 Occupational Health (Routine)

This attribute accounts for radiological exposures to workers during normal facility operations. Currently, surveillance is required by technical specifications for the hydrogen recombiners. For some plants, the recombiners are located inside containment. For such plants, during required surveillance and routine maintenance, workers who are in close proximity to the recombiners are exposed at an average rate of 10 mrem/hr (PWRs) and 20 mrem/hr (BWRs) [4]. A relaxation or deletion of the requirement would result in a dose savings to licensees.

According to industry estimates [8], it costs approximately $36,000 per year per reactor to operate and maintain a typical post-LOCA hydrogen recombiner system. Although this estimate is for a BWR, it is expected that costs for PWRs are similar. Of the $36,000, $14,000 is attributed to surveillance and maintenance. Assuming that one-fourth of this cost is directly attributed to time and labor spent in proximity to the recombiners, an estimate of dose savings can be derived. Using a cost of $3,500 for maintenance and surveillance, and an average industry labor rate of $80/hour, the resultant yearly exposure time is 44 hours. Thus, the dose per PWR is estimated to be 0.44 person-rem, and 0.88 person-rem for BWRs. The dose savings over 35 years, using the dollar per person-rem conversion factor of $2,000, would be $11,500 for each PWR and $23,000 for each BWR.

3.4.2.2.5 Offsite Property

The Offsite Property cost due to this relaxation was calculated consistent with the methodology described in Section 5.7.5 of [4]. From NUREG/CR-6349 [10], the offsite property consequences are about 6 percent of the magnitude of the public health costs for late containment failure for Peach Bottom. Thus, the Offsite Property cost savings is estimated to be -$1,300 per plant for Approach 1.

Since Approach 3 does not alter the recombiner requirements for BWRs with Mark I and Mark II containments, the numerical value for Offsite Property costs is zero.

3.4.2.2.6 Onsite Property

There is no change in the Onsite Costs, when comparing Approach 1 to the base case (Approach 4) since the onsite damage from the accident would have occurred in any event. This is also the case for Approach 3.

3.4.2.2.7 Industry Implementation

This attribute is an impact which accounts for the projected net economic effect on the affected licensees to install or implement mandated changes. Approach 1 will eliminate the requirement to maintain hydrogen recombiners. Since the recombiners will no longer be required, licensees may remove them permanently from service. Licensees could abandon the equipment in place, or permanently remove it. If licensees choose to remove the equipment, they will incur costs associated with the removal and radioactive waste disposal. However, if licensees choose to abandon the equipment in place, there will be some costs associated with instrumentation changes or deletions. For the purposes of this regulatory analysis it is assumed that an average of $10,000 per plant will be spent for the above implementation.

The relaxation in Approach 1 is likely to lead to a technical specification change. It will be to licensees' advantage to amend their technical specifications (remove the technical specification associated with recombiners); therefore, licensees will likely incur a cost for preparing and submitting a license amendment request. According to NUREG/CR-4627 [12], it costs approximately $28,000 (adjusted to 2002 dollars) to prepare a typical uncomplicated technical specification amendment request. Since it is likely that licensees will submit one license amendment request that will cover both the monitors and the recombiners, only half of the cost ($14,000) for the amendment is considered in this portion of the Value-Impact analysis. See Section 3.4.1.1.7 for inclusion of the remaining half of this cost.

3.4.2.2.8 Industry Operation

This attribute is an impact which measures the projected net economic effect due to routine and recurring activities required by the proposed action on all affected licensees. According to industry estimates [8], it costs approximately $36,000 per year per reactor to operate and maintain a typical post-LOCA hydrogen recombiner system. Although this estimate is for a BWR, it is expected that costs for PWRs are similar. Approach 1 will eliminate the requirement to maintain hydrogen recombiners. Therefore, a plant could expect annual savings of $36,000, or $470,000 over the remaining life assumed by this analysis.

3.4.2.2.9 NRC Implementation

Approach 1 will necessitate a rulemaking as well as revision to or development of regulatory guidance. The costs associated with the development of the rulemaking and associated guidance are sunk costs, and not considered by this regulatory analysis.

Because Approach 1 involves a deletion of a requirement, license amendments are expected on the part of the licensees, i.e., licensees will request an amendment to delete requirements associated with operation and surveillance of the recombiners. Therefore, the NRC will incur costs associated with review and approval of the amendment requests. According to NUREG/CR-4627 [12], it costs approximately $17,000 (adjusted to 2002 dollars) to review a typical uncomplicated technical specification amendment request. This cost includes preparation of a generic communication and model technical specification change. As was indicated in Section 3.4.1.1.9, the technical specification amendment request for recombiners is likely to be combined with the amendment request for the monitors. Therefore, $8,500 is assumed for this portion of the Value-Impact.

3.4.2.2.10 NRC Operation

This attribute is an impact which measures the projected net economic effect on the NRC after the proposed action is implemented. As a result of the proposed action, there will be a slight reduction in the effort during inspections. This reduction is expected to be small, and therefore will not be quantified for the purposes of this analysis.

3.4.2.2.11 Other Attribute Considerations

For completeness, the remaining attributes that make up the full set [4] are addressed here. Several – Safeguards and Security, Antitrust, Environmental, General Public, Improvement in Knowledge, and Other Government – have no bearing on this regulatory analysis and therefore are not discussed further. A discussion follows for the remaining one, Regulatory Efficiency. One of the major motivations for this rulemaking is to reduce unnecessary regulatory burden on both the industry and the NRC. This reduction in unnecessary regulatory burden results in a more efficient regulatory framework and refocuses resources on more risk significant activities.

4. Presentation of Results

4.1 Results for Monitors

Table 4.1 presents the “hydrogen monitor” results comparing Approach 1 (Option 1 from SECY-01-0162 [2]) to Approach 4 (the “No Change to Current Requirements, baseline Approach”) *for all BWRs and PWRs*. The Value-Impact indicates that Approach 1 is cost-beneficial, even when considering uncertainties. The Industry Value-Impact – the “per unit” Value-Impact times 103 units – is about $53M. There would be a slight adjustment to these numbers for BWRs with Mark I and Mark II containments in that the relaxation requirements for oxygen monitors should be taken into account. This impact is considered small and well within the uncertainties of the analysis.

Table 4.1 Results for Monitors in Approach 1 for All Plants

Quantitative Attribute			Present Value Estimate ($)
Health (value)	Public	Accident	0
		Routine	0
	Occupational	Accident	0
		Routine	0
Property (value)	Offsite		0
	Onsite		0
Industry (impact)	Implementation		70,000 + 40,000 + 14,000
	Operation		-650,000
NRC (impact)	Implementation		8,500
	Operation		0
NET Value (Sum)			517,000

From Table 4.1, the Value-Impact is calculated to be {(0)- (70,000+40,000+14,000+8,500-650,000)}= $517,500/plant, or about $520,000/plant.

The uncertainties for this evaluation are driven by the uncertainty in the result for Industry Operation. Only those uncertainties that would significantly reduce the magnitude of the result given, namely $650,000/plant, could have an impact on the conclusion for Approach 1. Elements of this uncertainty include: (1) the assumption that plant will obtain a life-extension of 20 years and (2) the assumption that the typical number used for operational savings per year provided in reference [8] is too large. If the assumption is made that there will be no license renewal and that the smallest magnitude number for operations savings is used (15 years of remaining life vs. 35 years or $40,000 per year vs. $50,000 per year) then the Industry Operation amount is $371,000. Even this number is large relative to other numbers in Table 4.1.

Another uncertainty relates to Approach 4, the no action reference case. The Value-Impact assessment described above does not consider the equipment replacement costs associated over 35 years of maintaining the status quo. It is assumed here that, if the Commission took no action, licensees would request exemptions, as was the case for Oconee [13]. This would be the less costly alternative to doing nothing and thus incurring the higher multimillion-dollar costs associated equipment replacement. Industry costs for an exemption are about $30,000, while NRC review of the exemption would run about $10,000. While these costs are not insignificant, they do not alter the conclusions of this regulatory analysis. Additionally, current Commission practice is to address generic issues through the rulemaking process. The rulemaking process vs. individual exemption process allows for greater public involvement, thereby increasing public confidence. Also, the rulemaking option would eliminate a non risk-significant requirement, and at the same time, would provide relief from unnecessary regulatory burden.

Thus, while there is some uncertainty in this analysis, it does not adversely affect the overall conclusion that Approach 1 is viable for all plants.

4.2 Results for Recombiners

Table 4.2 presents the “recombiner” results comparing Approach 1 (Option 1 from SECY-01-0162 [2]) compared to Approach 4 (the “No Change to Current Requirements, baseline

"Approach") *for all BWRs with Mark I or Mark II containments.* The Value-Impact indicates that Approach 1 is cost-beneficial, even when considering uncertainties. The Industry Value-Impact – the "per unit" Value-Impact times 30 units – is about $13M.

Table 4.2 Results for Recombiners in Approach 1 for Mark I and II Containments

Quantitative Attribute			Present Value Estimate ($)
Health (value)	Public	Accident	-21,300
		Routine	0
	Occupational	Accident	0
		Routine	23,000
Property (value)	Offsite		-1,300
	Onsite		0
Industry (impact)	Implementation		10,000 + 14,000
	Operation		-470,000
NRC (impact)	Implementation		8,500
	Operation		0
NET Value (Sum)			438,000

From Table 4.2, the Value-Impact is calculated to be {(-21,320+23,000 –1,300)-(10,000+14,000+8,500-470,000)} = $437,900/plant, or about $438,000/plant.

The uncertainties for this evaluation can be considered in two parts: uncertainties associated with the Values (Public and Occupational Health) and with the Impacts (NRC and Industry).

As was discussed in Section 3.4.2.1, value for the increased risk due to the relaxation is conservative, that is, the magnitude of the value is expected to be less. Using a less conservative value for Public-Accident, would make the "Value" portion of the equation even more positive, thereby further supporting Approach 1. Even if the Occupational-Routine contribution was zero, the total "Value" would be a relatively small, although a negative number. Thus, considering the uncertainties associated with the "Value" portion – that portion of the Value-Impact that focuses on protecting health and safety – the staff concludes that the result is either positive or negative but small, both in an absolute sense and relative to the results for the Impacts.

If the uncertainties for the "Impacts" are large and positive in sign, these uncertainties might challenge the conclusion that Approach 1 is cost-beneficial. Only if the uncertainties in the (positive) costs for NRC and Industry implementation are large can this happen (the result for Industry Operation is a best-estimate). If the amounts for NRC and Industry Implementation are doubled, the total Impact is still relatively large and negative, thus yielding an overall positive Value-Impact for Approach 1.

Even if the uncertainties are large, they do not adversely affect the overall conclusion that Approach 1 is viable for BWRs with Mark I or Mark II containments.

Approach 3, discussed in Section 2.3, also addresses recombiners, but is limited to plants with Mark I or Mark II containments. For these plants, Approach 3 would leave the recombiner requirements intact. Considering the recombiner issue for these plants then, the Value-Impact would be no different from doing nothing (Approach 4) while the Value-Impact from Approach 1 is sizable and positive. Thus, Approach 3 is not an attractive option from a Value-Impact perspective.

In Section 2.3, a variation of Approach 3 was addressed which retained the recombiners but relaxed the requirements for maintaining and operating them. The BWR Owners' Group estimates [8] that the annual cost savings of at least $25K could be expected if the recombiners were reclassified as non-safety. This equates to -$326K "Impact" over the life of the plant. Comparing this number to the equivalent for Approach 1, namely -$470K (note "Public-Accident" Value in Table 4.2), yields the conclusion that, while this variation on Approach 3 might be attractive, its Value-Impact is less than that of Approach 1 (The absolute values of the other attributes in the Value-Impact equation are smaller by at least an order of magnitude.)

Table 4.3 Results for Recombiners in Approach 1 for PWRs and Mark III Containments

Quantitative Attribute			Present Value Estimate ($)
Health (value)	Public	Accident	0
		Routine	0
	Occupational	Accident	0
		Routine	12,100[1]
Property (value)	Offsite		0
	Onsite		0
Industry (impact)	Implementation		10,000 + 14,000
	Operation		-470,000
NRC (impact)	Implementation		8,500
	Operation		0
NET Value (Sum)			449,600

[1]The value $12,100 was calculated based on 69 PWRs x $11,500 + 4 Mark III's x $23,000, then averaged over 73 plants.

Table 4.3 presents the "recombiner" results comparing Approach 1 (Option 1 from SECY-01-0162 [2]) compared to Approach 4 (the "No Change to Current Requirements, baseline "Approach") *for all BWRs with Mark III containments and all PWRs*. The Value-Impact indicates that Approach 1 is cost-beneficial, even when considering uncertainties. The Industry Value-Impact – the "per unit" Value-Impact times 73 units – is about $33M. From Table 4.3, the Value-Impact is calculated to be {(12,100)- (10,000+14,000+8,500-470,000)}= $449,600/plant, or about $450,000/plant.

The uncertainties for this evaluation can also be considered in two parts: uncertainties associated with the Values (Public and Occupational Health) and with the Impacts (NRC and Industry).

The only way that uncertainties in the Value portion can adversely impact the position that Approach 1 is viable is for the benefit of reducing the occupational routine value be reevaluated as zero. Thus, considering this uncertainty associated with the "Value" portion – that portion of the Value-Impact that focuses on protecting health and safety – the staff concludes that the result is positive but small, both in an absolute sense and relative to the results for the Impacts.

If the uncertainties for the “Impacts” are large and positive in sign, these uncertainties might challenge the conclusion that Approach 1 is cost-beneficial. Only if the uncertainties in the (positive) costs for NRC and Industry implementation are large can this happen (the result for Industry Operation is a best-estimate). If the amounts for NRC and Industry Implementation are doubled, the total Impact is still relatively large and negative, thus yielding an overall positive Value-Impact for Approach 1.

While the uncertainties might be large, they do not adversely affect the overall conclusion that Approach 1 is viable for BWRs with Mark III containments and all PWRs.

5. Decision Rationale

The conclusion drawn from this regulatory analysis is that the regulatory relaxation proposed as Approach 1 (Option 1 of SECY-01-0162) is appropriate from an overall safety and a Value-Impact perspective. The basic criteria for this determination is that the relaxation meets two specific conditions:

- the public health and safety and the common defense and security would continue to be adequately protected
- the cost savings attributed to the action would be substantial enough to justify taking action.

The risk and regulatory insights described in this regulatory analysis show that these rulemaking actions either do not increase risk or only increase risk slightly, such that there is virtually no change in the conditions for assuring that the public health and safety is adequately protected.

In addition, this analysis shows that the savings to the NRC and industry far outweigh the costs inherent in the action itself.

The Value-Impact demonstrates that the benefits, mainly in terms of relief from regulatory burden, far outweigh the small increase in risk for BWRs with Mark I and Mark II containments and far outweigh the essentially zero increase in risk for the PWRs and the BWRs with Mark III containments.

6. Implementation

The implementation of this action will be consistent with the schedule for the rulemaking provided in SECY-01-0162.

7. References

1. SECY-00-198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)," September 2000.

2. SECY-01-162, "Staff Plans for Proceeding with the Risk-Informed Alternative to the Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors in 10 CFR 50.44," August 2001.

3. "Regulatory Analysis Guidelines of the U.S. NRC," NUREG/BR-0058, Rev. 3, July 2000.

4. "Regulatory Analysis Technical Evaluation Handbook," NUREG/BR-0184, January 1997.

5. Attachment 2 to SECY-00-0198, "Feasibility Study for a Risk-Informed Alternative to 10 CFR 50.44," August 2000.

6. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and following an Accident," Regulatory Guide 1.97, Revision 3, May 1983.

7. Attachment 1 to SECY-00-0198, "Framework For Risk-Informed Changes to the Technical Requirements of 10 CFR 50, Draft, Revision 2," August 2000.

8. "Regulatory Relaxation For the H_2/O_2 Monitors and Combustible Gas Control System," BWR Owner's Group, July 2001.

9. PRM-50-68, 10 CFR Part 50, Bob Christie; Receipt of Petition for Rulemaking. 65 FR 1829. January 12, 2000.

10. Mubayi, V. et al., "Cost-Benefit Considerations in Regulatory Analysis," NUREG/CR-6349, BNL, October 1995.

11. *Applicant's Environmental Report - Operating License Renewal Stage, Turkey Point Units 3 & 4*. Florida City, FL, September 2000.

12. "Generic Cost Estimates, Abstracts from Generic Studies for Use in Preparing Regulatory Impact Analyses," NUREG/CR-4627, including Revs. 1 and 2, February 1992.

13. "Oconee Nuclear Station, Units 1, 2, and 3 RE: Exemption from the Requirements of Hydrogen Control Requirements of 10 CFR 50.44 .....," Letter to Mr. W.R. McCollum, Jr., Duke Energy Corporation, from Mr. David E. LaBarge, US NRC, dated July 17, 2001.

Appendix A

RISK-BENEFIT ANALYSIS OF COMBUSTIBLE GAS CONTROL FOR BWRS WITH MARK I CONTAINMENT

APPENDIX A

RISK-BENEFIT ANALYSIS OF COMBUSTIBLE GAS CONTROL FOR BWRS WITH MARK I CONTAINMENT

A.1 Introduction

In BWRs with Mark I containment, the containment atmosphere is normally maintained by nitrogen at a low concentration of oxygen, rendering it inert to combustion under most circumstances. Therefore, the only credible pathway leading to combustion in the containment is the long-term generation of hydrogen and oxygen by radiolysis in the suppression pool. After sufficient radiolysis has taken place, the concentration of oxygen in the containment may rise to a sufficiently high level (5 percent or greater) to de-inert the atmosphere, thus making combustion events possible. The radiolysis process is sensitive to such factors as accident timing; amount of liquid-phase iodine in the suppression pool; and the concentration of hydrogen in the containment atmosphere. De-inerting of containment is calculated in [A.1] to occur in about 3.6 days for conditions in which liquid-phase iodine represents 30 percent of the total core inventory, and would shorten for postulated conditions in which liquid-phase iodine approaches 75-100 percent of initial core inventory. However, the analysis did not take credit for the concentration of hydrogen in the containment atmosphere, which has been shown to have a strong effect on lengthening the time to de-inerting [A.6].

A.2 Basic Methodology

The risk-benefit associated with combustible gas control may be calculated using the formula:

$$\Delta R = C\, Z\, f_{CD} \sum_i \left(p_{i,base} - p_{i,sens} \right) D_i \qquad \text{(A.1)}$$

where

ΔR	=	net risk-benefit associated with combustible gas control ($);
C	=	effective number of years from the present over which to calculate the risk-benefit (years) (e.g., 13.05 years for a 35-year period calculated at a 7 percent discount rate, the average remaining lifetime of all U. S. reactors of General Electric design (including 20-year license extension) according to [A.2]);
Z	=	valuation factor for offsite dose consequence ($/person-rem) (a value of $2000/person-rem calculated within a 50-mile radius is recommended by [A.2]);
f_{CD}	=	total core damage frequency for risk-significant sequences (events/reactor-year);

$p_{i,base}$ = conditional probability of containment failure mode or release class *i* in the baseline case without combustible gas control;

$p_{i,sens}$ = conditional probability of containment failure mode or release class *i* in the sensitivity case with combustible gas control; and

D_i = offsite dose consequence associated with containment failure mode or release class *i* (person-rem/event).

The three main elements of data required are thus the frequency of risk-significant core damage events; the conditional probabilities of containment failure; and the offsite health consequences of containment failure. For this study, Peach Bottom Unit 2 is used as a reference plant, since it was used as the reference for NUREG/CR-4551 [A.3] and therefore has the most available data. Where possible, data from the Peach Bottom IPE [A.4] was used as well.

A.3 Risk-Significant Event Frequency

Risk-significant sequences for this study are represented by all sequences in which the accident progresses past the late time frame (1-3 days) with an intact containment. In case of a pre-existing, early, or late containment failure by other means, the radiolysis issue is rendered irrelevant. Moreover, sequences leading to controlled containment venting are not included, since it is assumed that the releases and consequences resulting from the earlier venting will themselves be much greater than those resulting from the very late containment rupture induced by combustion of gases produced by radiolysis.

From the IPE, the total core damage frequency due to internal events is about 5.53e-6 per reactor-year, of which 46.4 percent (page 4.6-30 of [A.4]) result in a late intact containment. Therefore, the frequency of risk-significant sequences for internal initiators is 2.57e-6 per reactor-year.

NUREG/CR-4551 [A.3] is used at present as having the most usable data for Peach Bottom on external event initiators. From Figure 2.5-9 in that document, the frequency of core damage due to fires that result in a late intact containment is 4.69e-6 per reactor-year (i.e., about 24 percent of the total fire CDF of 1.98e-5 per reactor-year). Figures 2.5-11(a, b) in [A.3] show that there is zero probability of seismic core damage sequences resulting in a late intact containment.

These frequencies are summarized in Table A.1.

A.4 Containment Failure Probabilities

The sequences in the baseline case, by definition, all have late intact containment. For the sensitivity case, it is assumed that the lack of combustible gas control will in all of the same circumstances result in a very late, catastrophic failure of the drywell. The resulting containment response matrix is shown in Table A.2.

A.5 Consequences

From NUREG/CR-4551, representative source terms are available for core damage sequences leading to an intact containment, for both internally and externally initiated sequences. These source terms are shown in Table A.3. Comparing to Tables 3.4-4 and 3.4-8 in [A.3], it can be seen that these correspond most closely to release classes PB-17-1 (for internally initiated events) and PBF-19-1 (for fires). The resulting consequences, from Tables 4.3-1 and 4.3-2 in [A.3], are 52.2 person-rem/event and 62.9 person-rem/event, respectively. Consequences are summarized in Table A.4.

Source terms corresponding to a very late catastrophic rupture of the containment are unavailable in NUREG/CR-4551; all containment failures considered there occur within about 40,000 seconds (11 hours) of scram. Instead, it is proposed for now to use the source terms for late containment failure, typical values of which are shown in Table A.3 (taken from, e.g., Figure 3.3-15 in [A.3]). These source terms are approximately represented by release classes PB-1-1 (for internal events) and PBF-1-1 (for fires), with consequences of 1.82e5 person-rem/event and 7.45e4 person-rem/event, respectively.

A.6 Results

Using Equation (A.1), the risk-benefit associated with combustible gas control for Peach Bottom can now be calculated as:

$$\Delta R_{internal} = (2.57\times 10^{-6})(13.05)(\$2000)\left[(1.0)(1.82\times 10^{5} - 52.2)\right]$$
$$= (13.05)(\$2000)(0.468)$$
$$= \$12{,}210. \qquad \text{(A.2)}$$

$$\Delta R_{fires} = (4.69\times 10^{-6})(13.05)(\$2000)\left[(1.0)(7.45\times 10^{4} - 62.9)\right]$$
$$= (13.05)(\$2000)(0.349)$$
$$= \$9110. \qquad \text{(A.3)}$$

$$\Delta R_{seismic} = 0. \qquad \textbf{(A.4)}$$

$$\Delta R_{total} = \Delta R_{int} + \Delta R_{fires} + \Delta R_{seismic}$$
$$\boxed{= \$21{,}320.} \qquad \text{(A.5)}$$

These results are also summarized in Table A.5.

A.7 Conclusions

Using available information from the Peach Bottom IPE and NUREG/CR-1150, a bounding risk-benefit of about $21,320 has been found for control of combustible gases and oxygen produced during radiolysis. This is a conservative estimate, given that the actual source term and consequences for very late containment failure (several days after scram) are likely to be significantly lower than those for late containment failure (less than 12 hours after scram), which were used in the calculation. Nevertheless, the resulting benefit is relatively small. This is largely attributable to the fact that consequences for such late failure times are relatively small.

Note that this analysis has not included offsite economic consequences of the proposed action. In view of past consequence calculations, the offsite economic consequences are generally of similar magnitude to the offsite health consequences. In [A.5] (Table 4-6), it is in fact seen that the conditional offsite health and property consequences for late containment failure (PB-01-1) are 2.05e5 person-rem and $2.40e7, respectively. Using a conversion factor of $2000/person-rem, it is seen that property costs are only about 6 percent of the health costs. If the result of the present analysis were to be increased by the same proportion to include property costs, then the total benefit would become $22,600.

A.8 <u>References</u>

A.1. BWR Owners' Group, "Licensing Topical Report: Regulatory Relaxations for the H_2/O_2 Monitors and Combustible Gas Control Systems", NEDO-33003, General Electric Company, July 2001.

A.2. U. S. NRC Office of Regulatory Research, "Regulatory Analysis Technical Evaluation Handbook", NUREG/BR-0184, U. S. Nuclear Regulatory Commission, January 1997.

A.3. A. C. Payne, R. J. Breeding, et al., "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2 – Main Report", Vol. 4, Rev. 1, Part 1, NUREG/CR-4551, December 1990.

A.4. G. J. Beck et al., "Individual Plant Examination, Peach Bottom Atomic Power Station Units 2 & 3", Philadelphia Electric Co., August 1992.

A.5. V. Mubayi, V. Sailor, et al., "Cost-Benefit Considerations in Regulatory Analysis", NUREG/CR-6349, Brookhaven National Laboratory, October 1995.

A.6. K. I. Parczewski and V. Benarroya, "Generation of Hydrogen and Oxygen by Radiolytic Decomposition of Water in Some BWRs", presented at Joint ASME/ANS Nuclear Engineering Conference, Portland, Oregon, August 5-8, 1984.

Table A.1 Event Frequencies for Peach Bottom Unit 2

Initiator Category	Total CDF (events/year)	CDF with Late Intact Containment (events/year)	Conditional Probability of Late Intact Containment
Internal Events[1]	5.53e-6	2.57e-6	0.46
Fires[2]	1.98e-5	4.69e-6	0.24
Seismic Events[2]	7.52e-5	0	0
Total	1.01e-4	7.26e-6	

1 Source: IPE [A.4].
2 Source: NUREG/CR-4551 [A.3].

Table A.2 Containment Matrix for Peach Bottom Unit 2 (Sequences with Late Intact Containment in Baseline Case)

Case	Conditional Probability of No Containment Failure	Conditional Probability of Very Late Catastrophic Containment Rupture
Baseline (without combustible gas control)	0.0	1.0
Sensitivity (with combustible gas control)	1.0	0.0

Table A.3 Source Terms for Peach Bottom Unit 2 (from NUREG/CR-4551 [A.3])

Containment Failure Mode or Release Class	Xe	I	Cs	Te	Ba	Sr	La
No CF	2e-3	1e-4	1e-8	1e-9	1e-9		1e-10
PB-17-1	4e-3	3e-6	6e-9	2e-9	2e-9	2e-9	1e-10
PBF-19-1	3e-3	5e-6	4e-9	2e-9	7e-10	8e-10	6e-11
Late CF	1.0	1e-2	5e-4			5e-5	5e-6
PB-1-1	0.95	1e-2	7e-4	4e-4	6e-5	6e-5	6e-6
PBF-1-1	0.95	1e-2	1e-4	6e-5	3e-5	3e-5	2e-6

Table A.4 Consequences for Peach Bottom Unit 2 Release Classes (from NUREG/CR-4551 [A.3])

Release Class	Description	Conditional Offsite Health Consequence (person-rem/event, 50-mile radius)
PB-17-1	No CF (Internal Events)	5.22e1
PBF-19-1	No CF (Fires)	6.29e1
PB-1-1	Late CF (Internal Events)	1.82e5
PBF-1-1	Late CF (Fires)	7.45e4

Table A.5 Summary of Risk-Benefit Results for Combustible Gas Control at Peach Bottom Unit 2

Initiator Category	Net Change in Consequence (person-rem/year)	Net Risk-Benefit ($)
Internal Events	0.468	$12,210
Fires	0.349	$9110
Seismic Events	0	$0
Total	**0.817**	**$21,320**

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CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT

A. INTRODUCTION

The NRC has issued a revision to Section 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,” which is an amendment to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.” This regulation is applicable to all reactor construction permits or operating licenses under this part, except for those facilities for which the certifications required under §50.82(a)(1) have been submitted, and to all reactor design approvals, design certifications, combined licenses or manufacturing licenses under 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.” This regulatory guide was developed to describe methods that are acceptable to the NRC staff for implementing the revised Section 50.44.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget, approval number 3150-0011. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

B. DISCUSSION

Section 50.44 provides requirements for the mitigation of combustible gas generated by a beyond-design-basis accident. In existing light-water reactors, the principal combustible gas is hydrogen.

In an accident more severe than the design-basis loss-of-coolant accident (LOCA), combustible gas is predominately generated within the containment as a result of:

1. Fuel clad-coolant reaction between the fuel cladding and the reactor coolant, and
2. Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

If a sufficient amount of combustible gas is generated, it may react with oxygen present in the containment at a rate rapid enough to lead to the breaching of the containment or a leakage rate in excess of technical specification limits. Additionally, damage to systems and components essential to continued control of the post-accident conditions could occur.

In SECY-00-0198, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) And Recommendations on Risk-informed Changes to 10 CFR 50.44 (Combustible Gas Control),” dated September 14, 2000 (Ref. 1), the NRC staff recommended changes to 10 CFR 50.44 that reflect the position that only combustible gas generated by a beyond-design-basis accident is a risk-significant threat to containment integrity. Based on those recommendations, the recent revision to 10 CFR 50.44 eliminates requirements that pertain to only design-basis LOCAs.

Attachment 2 to SECY-00-198 (Ref. 1) used the framework described in Attachment 1 to the paper with risk insights from NUREG-1150 (Ref. 2) and the integrated plant evaluation programs to evaluate the requirements in 10 CFR 50.44. It was noted in Attachment 2 that containment types that rely on pressure suppression concepts (i.e., ice baskets or water pools) to condense the steam from a design-basis LOCA have smaller containment volumes, and in some cases lower design pressures, than pressurized water reactor (PWR) large-volume or subatmospheric containments. Consequently, the smaller volumes and lower design pressures associated with pressure suppression containment designs make them more vulnerable to combustible gas deflagrations during degraded core accidents because the pressure loads could cause structural failure of the containment. Also, because of the smaller volume of these containments, detonable mixtures could be formed. A detonation would impose a dynamic pressure load on the containment structure that could be more severe than the static load from an equivalent deflagration. However, the staff noted in SECY-00-0198 that the risk of early containment failure from combustible gas combustion in these types of containments can be limited by the use of mitigative features: (1) inerting in Mark I and II containments and (2) using igniter systems in Mark III and ice condenser containments. As a result, the revised Section 50.44 has the following requirements:

1. All boiling water reactor (BWR) Mark I and II type containments must be inerted. By maintaining an oxygen-deficient atmosphere, combustible gas combustion that could threaten containment integrity is prevented.

2. All BWRs with Mark III type containments and all PWRs with ice condenser type containments must have the capability to control combustible gas generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity. The deliberate ignition systems provided to meet this existing combustible gas source term are capable of safely accommodating even greater amounts of combustible gas associated with even more severe core melt sequences that fail the reactor vessel and involve molten core-concrete interaction. Deliberate ignition systems, if available, generally consume the combustible gas before it reaches concentrations that can be detrimental to containment integrity.

3. For all applicants for and holders of a water-cooled reactor construction permit or operating license under 10 CFR Part 50, and to all applicants for a light-water reactor design approval, or design certification, or combined license under 10 CFR Part 52 that are docketed after the effective date of the rule, the following requirements apply. All containments must have an inerted atmosphere or limit combustible gas concentrations in containment during and following an accident that releases an equivalent amount of combustible gas as would be generated from a 100% fuel-clad coolant reaction, uniformly distributed, to less than 10% (by volume) and must maintain containment structural integrity. The requirements of this paragraph apply only to water-cooled reactor designs with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of [EFFECTIVE DATE OF RULE].

4. For all construction permits and operating licenses under this part, and to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 of this chapter, for non water-cooled reactors and water-cooled reactors which do not fall within the description in paragraph 3. above, any of which are issued after the effective date of the rule, applications subject to this paragraph must include:

    1. Information addressing whether accidents involving combustible gases are technically relevant for their design, and

    2. If accidents involving combustible gases are found to be technically relevant, information demonstrating that the safety impacts of combustible gases during risk-significant accidents have been addressed to ensure adequate protection of public health and safety and common defense and security.

The combustible gas control systems, the atmosphere mixing systems, and the provisions for measuring and sampling that are required by Section 50.44 are risk-significant as they have the ability to mitigate the risk associated with combustible gas generation caused by significant beyond-design-basis accidents. The recommended treatments for those systems are delineated in the Regulatory Position.

C. REGULATORY POSITION

1. COMBUSTIBLE GAS CONTROL SYSTEMS

The following design guidance is applicable to combustible gas control systems installed to mitigate the risk associated with combustible gas generation due to beyond design basis accidents. Structures, systems, and components (SSCs) installed to mitigate the hazard from the generation of combustible gas in containment should be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Equipment survivability expectations under severe accident conditions should consider the circumstances of applicable initiating events (such as station blackout[1] or earthquakes) and the environment (including pressure, temperature, and radiation) in which the equipment is relied upon to function. This guidance was contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993 (Ref. 3).

The required system performance criteria will be based on the results of design-specific reviews that include probabilistic risk-assessment as required by 10 CFR 52.47(a)(v). Because these requirements address beyond-design-basis combustible gas control, SSCs provided to meet these requirements need not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirements of Appendix B to 10 CFR Part 50, and redundancy/diversity requirements of Appendix A to 10 CFR Part 50. Guidance such as that found in Appendices A and B of Regulatory Guide 1.155 (Ref. 4) is appropriate for equipment used to mitigate the consequences of severe accidents. This guidance was used to review the design of evolutionary and passive plant designs as documented in NUREG-1462 (Ref. 5), NUREG-1503 (Ref. 6), and NUREG-1512 (Ref. 7).

The combustible gas control systems in all BWRs with Mark III-type containments and all PWRs with ice condenser type containments must meet the requirements in the Revised Section 50.44. The staff considers that the combustible gas control systems installed and approved by the NRC as of the effective date of the rule are acceptable without modification.

2. OXYGEN AND HYDROGEN MONITORS

2.1 Hydrogen Monitors

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[1] The revised Section 50.44 does not require the deliberate ignition systems used by BWRs with Mark III type containments and PWRs with ice condenser type containments to be available during station blackout events. The deliberate ignition systems should be available upon restoration of power. Additional guidance concerning the availability of deliberate ignition systems during station blackout sequences is being developed as part of the staff's review of Generic Safety Issue 189, “Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident.”

The Revised Section 50.44 requires that equipment be provided for monitoring hydrogen in the containment. The equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a beyond-design-basis accident for accident management, including emergency planning. Safety-related hydrogen monitoring systems installed and approved by the NRC prior to the effective date of the rule are sufficient to meet these criteria. Non-safety-related commercial grade hydrogen monitors can also be used to meet these criteria if they:

1. Comply with the Category 3 design and qualification criteria of Regulatory Guide 1.97 (Ref. 8) for monitors used as diagnostic or backup indicators.

2. Comply with the Category 2 power source design and qualification criteria as specified in Table 1 of Regulatory Guide 1.97 (Ref. 8).

The above provisions can be met with a program based on compliance with a pre-specified, structured program of testing and calibration; alternatively, these items can be met with a less-prescriptive, performance-based approach to assurance of the hydrogen monitoring function. Such an approach is consistent with SECY-00-191, "High-Level Guidelines for Performance-Based Activities" (Ref. 9). Specifically, assurance of the reliability, availability, and capability of the hydrogen monitoring function can be derived through tracking actual reliability performance (including calibration) against targets established by the licensee based on the significance of this function, which is determined on a plant-specific basis. Thus, for hydrogen monitoring, it is acceptable to accomplish the functions of servicing, testing, and calibration within the maintenance rule program provided that applicable targets are established based on the functions of the hydrogen monitors delineated above.

Section 50.44 also requires that hydrogen monitors be functional. Functional requirements can be found in TMI Action Item II.F.1, Attachment 6, in NUREG-0737 (Ref. 10), which states that hydrogen monitors are to be functioning within 30 minutes of the initiation of safety injection. This requirement was imposed by confirmatory orders following the Three Mile Island Unit 2 accident. Since that requirement was issued, the staff has determined that 30 minutes can be overly burdensome. Through the "Confirmatory Order Modifying Post-TMI Requirements Pertaining to Containment Hydrogen Monitors for Arkansas Nuclear One, Units 1 and 2" (Ref. 11), dated September 28, 1998, the staff developed a method for licensees to adopt a risk-informed functional requirement in lieu of the 30-minute requirement. As described in the confirmatory order, an acceptable functional requirement would meet these requirements:

i. Procedures shall be established for ensuring that indication of hydrogen concentration in the containment atmosphere is available in a sufficiently timely manner to support the role of information in the Emergency Plan (and related procedures) and related activities such as guidance for the severe accident management plan.

ii. Hydrogen monitoring will be initiated on the basis of:

    (1) The appropriate priority for establishing indication of hydrogen concentration within containment in relation to other activities in the control room.

(2) The use of the indication of hydrogen concentration by decision makers for severe accident management and emergency response.

(3) Insights from experience or evaluation pertaining to possible scenarios that result in significant generation of hydrogen that would be indicative of core damage or a potential threat to the integrity of the containment building.

The NRC staff has found that adoption of this functional requirement by licensees results in the hydrogen monitors being functional within 90 minutes after the initiation of safety injection. This period of time includes equipment warm-up but not equipment calibration.

2.2 Oxygen Monitors

The Revised Section 50.44 requires that equipment be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. The revised rule requires the equipment for monitoring oxygen to be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a beyond design-basis accident for combustible gas control and accident management, including emergency planning. Existing oxygen monitoring systems approved by the NRC prior to the effective date of the rule are sufficient to meet this criterion. Non-safety-related oxygen monitors would also meet these criteria if they meet the Category 2 design and qualification criteria of Regulatory Guide 1.97 (Ref. 8) for monitors designated for indicating system operating status.

3. ATMOSPHERE MIXING SYSTEMS

The Revised Section 50.44 requires that all containments have a capability for ensuring a mixed atmosphere. This capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features that promote the free circulation of the atmosphere. All containment types should have an analysis of the effectiveness of the method used for providing a mixed atmosphere. This analysis should demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity.

Atmosphere mixing systems prevent local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single-failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power. The NRC staff considers atmosphere mixing systems installed and approved by the NRC as of the effective date of the rule to be acceptable without modification.

4. HYDROGEN GAS PRODUCTION

Materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

5. CONTAINMENT STRUCTURAL INTEGRITY

The Revised Section 50.44 requires that containment structural integrity be demonstrated by use of an analytical technique that is accepted by the NRC staff. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The following criteria of the ASME Boiler and Pressure Vessel Code provide an acceptable method for demonstrating that the requirements are met.

i. That steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subsubarticle NE - 3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required); and

ii. That concrete containments meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC - 3720, Factored Load Category, considering pressure and dead load alone.

As a minimum, the specific code requirements set forth for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant.

These criteria, while being removed from the existing regulations, are acceptable to the NRC staff for meeting the revised regulations. The acceptability of licensee analyses using the ASME Code criteria remains unaffected by this rulemaking.

REFERENCES

1. SECY-00-0198, “Status Report on Study of Risk-Informed Changes to The technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control),” September 14, 2000.[2]

2. USNRC, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, December 1990.[3]

3. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-water Reactor (ALWR) Designs,” USNRC, April 2, 1993.[1]

4. USNRC, “Station Blackout,” Regulatory Guide 1.155, August 1988.[3]

5. USNRC, “Final Safety Evaluation Report Related to the Certification of the Advance Boiling Reactor Design,” NUREG-1462, August 1994.[2]

6. USNRC, “Final Safety Evaluation Report Related to the Certification of the System 80+ Design, Docket No. 52-002,” NUREG-1503, July 1994.[2]

7. USNRC, “Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design, Docket No. 52-003,” NUREG-1512, September 1998.[2]

8. USNRC, “Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident,” Regulatory Guide 1.97, Revision 3, May 1983.[4]

9. USNRC, "High-Level Guidelines for Performance-Based Activities," SECY-00-0191, September 1, 2000.[1]

10. USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.[2]

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[1] Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike (first floor), Rockville, MD; the PDR’s mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or 1-(800)397-4209; fax (301)415-3548; e-mail <PDR@NRC.GOV>.

[3] Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; <http://www.ntis.gov/ordernow>, telephone (703)487-4650; . Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR’s mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or (800)397-4209; fax (301)415-3548; email is PDR@NRC.GOV.

[4] Single copies of regulatory guides, both active and draft, and draft NUREG documents may be obtained free of charge by writing the Reproduction and Distribution Services Section, OCIO, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to <DISTRIBUTION@NRC.GOV>. Active guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161; telephone (703)487-4650; online <http://www.ntis.gov/ordernow>. Copies of active and draft guides are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR’s mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or (800)397-4209; fax (301)415-3548; email <PDR@NRC.GOV>.

11. USNRC, “Confirmatory Order Modifying Post-TMI Requirements Pertaining to Containment Hydrogen Monitors for Arkansas Nuclear One, Units 1 and 2,” September 28, 1998.[1]

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this guide. The draft regulatory analysis prepared for the revision to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-water-Cooled Power Reactors," provides the regulatory basis for this guide and examines the costs and benefits for the rule as implemented by the guide. A copy of this regulatory analysis is available for inspection or copying for a fee in the NRC Public Document Room, located at 11555 Rockville Pike (first floor), Rockville, Maryland. This regulatory analysis is also available in the NRC's Electronic Reading Room, in the ADAMS system, under Accession Number ML021080807.

BACKFIT ANALYSIS

This regulatory guide was developed to describe a voluntary method that is acceptable to the NRC staff for complying with the requirements of 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-water-Cooled Power Reactors." Compliance with this regulatory guide is not a requirement, and a licensee may chose this or another way to achieve compliance with these rules. This 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 staff position; nor does it require the modification of or addition to systems, structures, components, or design of a facility, or the procedures or organization required to design, construct, or operate a facility.

ATTACHMENT 6: STANDARD REVIEW PLAN SECTION 6.2.5

DRAFT REVISION TO
STANDARD REVIEW PLAN - NUREG-0800
IN CONJUNCTION WITH RISK-INFORMED REVISION TO 50.44

SECTION 6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Plant Systems Branch (SPLB)

Secondary - None

I. AREAS OF REVIEW

10 CFR 50.44, “Combustible Gas in Containment,” is applicable to all power reactors. Regulatory Guide 1.7, Revision 3, “Control of Combustible Gas Concentrations in Containment” (Ref. 1), describes methods that are acceptable to the NRC staff for implementing 10 CFR 50.44.

Note: This SRP is primarily intended to cover new water-cooled reactor plant applications with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of [EFFECTIVE DATE OF RULE]. Guidance for a plant which had already received its operating license as of [EFFECTIVE DATE OF RULE], or for non-water-cooled reactor plants or water-cooled reactor plants that do not fall within the description above, may be found in Regulatory Guide 1.7, Revision 3.

SPLB reviews the information presented in the applicant's safety analysis report (SAR) or design control document (DCD) concerning the control of combustible gases in the containment following a beyond-design-basis accident involving 100% fuel clad-coolant reaction or postulated accident to ensure conformance with the requirements of General Design Criteria 5, 41, 42, and 43, and 10 CFR 50.44. Following an accident, hydrogen and oxygen may accumulate inside the containment.

After an accident, combustible gas is predominantly generated within the containment as a result of:

a. Fuel clad-coolant reaction between the fuel cladding and the reactor coolant.

b. Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

If a sufficient amount of combustible gas is generated, it may react with the oxygen present in the containment at a rate rapid enough to breach the containment or cause a leakage rate in excess of Technical Specification limits. Additionally, the associated pressure and temperature increase could damage systems and components essential to continued control of the post-accident conditions.

The SPLB review includes the following general areas:

1. The production and accumulation of combustible gases within the containment following a beyond design-basis accident.

2. The capability to mix the combustible gases with the containment atmosphere and prevent high concentrations of combustible gases in local areas.

3. The capability to monitor combustible gas concentrations within containment, and, for inerted containments, oxygen concentrations within containment.

4. The capability to reduce combustible gas concentrations within containment by suitable means, such as igniters.

The SPLB review specifically covers the following analyses and aspects of combustible gas control system designs:

1. Analysis of combustible gas (e.g., hydrogen, carbon monoxide, oxygen) production and accumulation within the containment following a beyond-design-basis accident.

2. Analysis of the functional capability of the systems or passive design features provided to mix the combustible gas within the containment.

3. Analysis of the functional capability of the systems provided to reduce combustible gas concentrations within the containment.

4. Analyses of the capability of systems or system components to withstand dynamic effects, such as transient differential pressures that would occur early in the blowdown phase of an accident.

5. Analyses of the consequences of single active component malfunctions, to meet GDC 41.

6. The quality classification of each system.

7. The seismic design classification of each system.

8. The results of qualification tests performed on system components to demonstrate functional capability.

9. The design provisions and proposed program (including Technical Specifications at the operating license (OL) or combined license (COL) stage of review) for periodic inservice inspection, operability testing, and leakage rate testing of each system or component.

10. The functional aspects of instrumentation provided to monitor system or system component performance.

At the construction permit (CP) or early site permit stage of review, the design of the systems provided for monitoring and controlling combustible gases within the containment may not be completely determined. In such cases, SPLB reviews the applicant's preliminary designs and statements of intent to comply with the acceptance criteria for such systems. At the OL or COL stage, SPLB reviews the final designs of these systems to verify that they meet the acceptance criteria detailed in subsection II of this SRP section. For design approvals and certifications, SPLB reviews the applicant's preliminary designs and statements of intent to comply with the acceptance criteria for such systems.

Review Interfaces

SPLB will coordinate other branch evaluations that interface with the overall review of combustible gas control as follows:

1. The Mechanical and Civil Engineering Branch (EMEB) will review seismic design and quality group classifications as part of its primary review responsibility for SRP Section 3.2.1 and SRP Section 3.2.2, respectively.

2. The Electrical and Instrumentation and Controls Branch (EEIB), as part of its primary review responsibility for SRP Section 7.5, will evaluate the actuation and control features of active components, including the hydrogen and oxygen monitors.

3. The EEIB, as part of its primary review responsibility for SRP Section 3.11, will evaluate the qualification test program for electric valve operators, fans, hydrogen/oxygen sampling or analyzing equipment, igniters, and sensing and actuation instrumentation of the plant protection system, located both inside and outside the reactor containment.

4. The Probabilistic Safety Assessment Branch (SPSB), as part of its primary review responsibility for SRP Section 12.3, will evaluate the accessibility of combustible gas control systems equipment under postulated accident conditions.

5. The Operating Reactor Improvements Program (RORP), as part of its primary review responsibility for SRP Section 16.0, will review, at the OL or COL stage of review, proposed Technical Specifications pertaining to the operability and leakage rate testing of systems and components.

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

II. ACCEPTANCE CRITERIA

SPLB acceptance criteria for the design of the systems provided for combustible gas control are the relevant requirements of 10 CFR Part 50, § 50.44, and General Design Criteria 5, 41, 42, and 43. The requirements are as follows:

1. 10 CFR Part 50, § 50.44, as it relates to BWR and PWR plants being designed to:

   a. accommodate hydrogen generation equivalent to a 100% fuel clad-coolant reaction,

   b. limit containment hydrogen concentration to no greater than 10%,

   c. have a capability for ensuring a mixed atmosphere during design-bases and significant beyond-design-bases accidents (a significant beyond-design-basis accident is an accident comparable to a degraded core accident at an operating (as of [EFFECTIVE DATE OF RULE]) light-water reactor in which a metal-water reaction occurs involving 100% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume)) ,

   d. provide containment-wide hydrogen control (such as igniters or inerting), if necessary, for certain severe accidents. Post-accident conditions should be such that an uncontrolled hydrogen/oxygen recombination would not take place in the containment, or the plant should withstand the consequences of uncontrolled hydrogen/oxygen recombination without loss of safety function or containment structural integrity.

2. General Design Criterion 5 as it relates to providing assurance that sharing of structures, systems and components important to safety among nuclear power units will not significantly impair their ability to perform their safety functions.

3. General Design Criterion 41 as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; systems being designed to suitable requirements, i.e., that there be suitable redundancy in components and features, and suitable interconnections to ensure that for either a loss of onsite or a loss of offsite power the system safety function can be accomplished, assuming a single failure; and systems being provided with suitable leak detection, isolation, and containment capability to ensure that system safety function can be accomplished.

4. General Design Criterion 42 as it relates to the design of the systems to permit appropriate periodic inspection of components to ensure the integrity and capability of the systems.

5. General Design Criterion 43 as it relates to the systems being designed to permit periodic testing to ensure system integrity, and the operability of the systems and active components.

Specific criteria necessary to meet the requirements of 10 CFR Part 50, § 50.44, and GDC 5, 41, 42 and 43, are as follows:

1. In meeting the requirements of 10 CFR Part 50, § 50.44, and GDC 41 to provide systems to control the concentration of hydrogen in the containment atmosphere, materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

2. In meeting the requirements of 10 CFR Part 50, § 50.44, and GDC 41 to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere, the applicant should demonstrate by analysis, for non-inerted containments, that the design can safely accommodate hydrogen generated by an equivalent of a 100% fuel clad-coolant reaction, while limiting containment hydrogen concentration, with the hydrogen uniformly distributed, to less than 10% (by volume), and while maintaining containment structural integrity.

3. In meeting the requirements of 10 CFR Part 50, § 50.44(c)(3), regarding equipment survivability, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity should perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel clad-coolant reaction including the environmental conditions created by activation of the combustible gas control system.

4. In meeting the requirements of 10 CFR Part 50, § 50.44, to provide the capability for ensuring a mixed atmosphere in the containment during design-bases and significant beyond-design-bases accidents, and of GDC 41 to provide systems as necessary to ensure that containment integrity is maintained, this capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features which promote the free circulation of the atmosphere. For all containment types, an analysis of the effectiveness of the method used for providing a mixed atmosphere should be provided. This analysis is acceptable if it shows that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity.

   Atmosphere mixing systems prevent local accumulation of combustible or detonable gases which could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power.

5. In meeting the requirements of 10 CFR Part 50, § 50.44, and GDC 41 regarding the functional capability of the combustible gas control systems to ensure that containment integrity is maintained, the design should meet the provisions of Regulatory Guide 1.7, Revision 3, section C.1.

6. To satisfy the design requirements of GDC 41:

a. Performance tests should be performed on system components, such as hydrogen igniters and combustible gas monitors. The tests should support the analyses of the functional capability of the equipment.

b. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room. The containment hydrogen and oxygen monitors should meet the provisions of Regulatory Guide 1.7, Revision 3, section C.2.

7. To satisfy the inspection and test requirements of GDC 41, 42 and 43, combustible gas control systems should be designed with provisions for periodic inservice inspection, operability testing, and leak rate testing of the systems or components.

8. In meeting the requirements of 10 CFR Part 50, § 50.44(c)(5), regarding containment structural integrity, an analysis must demonstrate containment structural integrity, using an analytical technique that is accepted by the NRC staff and including sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100% fuel clad-coolant reaction accompanied by combustible gas burning. Systems necessary to ensure containment integrity must also demonstrate the capability to perform their functions under these conditions. One acceptable analytical technique is a demonstration that specific criteria of the ASME Boiler and Pressure Vessel Code, described in Regulatory Guide 1.7, Revision 3, section C.5, are met.

9. In meeting the requirements of 10 CFR Part 50, § 50.44(c), and GDC 41 for the design and functional capability of the combustible gas control systems, preliminary system designs and statements of intent in the SAR are acceptable at the CP or early site

permit stage of review if the guidelines of Regulatory Guide 1.7, Revision 3, are endorsed.

III. REVIEW PROCEDURES

The procedures described below provide guidance for the detailed review of the combustible gas control systems. The reviewer selects and emphasizes material from this SRP section, as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of combustible gas control systems design common to a class of plants or by adopting the results of previous reviews of similar plants.

Upon request from the primary reviewer, other review branches will provide input for the areas of review stated in subsection I, above. The primary reviewer obtains and uses such input as required to ensure that this review procedure is complete.

The combustible gas control systems include systems for mixing the combustible gases, monitoring combustible gas concentrations, and reducing the combustible gas concentrations. In general, all of the combustible gas control systems should meet the design requirements outlined in subsection II. The system description and schematic drawings presented in the safety analysis report should be sufficiently detailed to permit judgments to be made regarding system acceptability.

1. SPLB determines that all potential, active mechanical failures and passive electrical failures have been identified and that no single failure would incapacitate an entire system.

2. SPLB compares the quality standards applied to the systems to the provisions of Regulatory Guide 1.7, Revision 3.

3. SPLB compares the seismic design classifications of the systems to the provisions of Regulatory Guide 1.7, Revision 3.

4. SPLB reviews the qualification testing of systems and components, to establish the functional capability of the equipment.

5. SPLB reviews the provisions made in the design of the systems and the program for periodic inservice inspection and operability testing of the systems or components. The inspections are reviewed with regard to the purpose of each inspection. The operability tests that will be conducted are reviewed with regard to what each test is intended to accomplish. Judgment and experience from previous reviews are used to determine the acceptability of the inspection and test program.

6. SPLB reviews the proposed technical specifications, for plants at the OL or COL stage of review, for the systems used to control and monitor combustible gas and oxygen concentrations in the containment to ensure that the requirements of 10 CFR 50.44 and General Design Criteria 5, 41, 42, and 43 are met.

7. SPLB reviews the capability to monitor system performance and control active components to be sure that control can be exercised over a system and that a malfunctioning system train or component can be isolated. The instrumentation provided

for this purpose should be redundant and should enable the operator to identify the malfunctioning system train or component.

8. SPLB reviews analyses of the functional capability of the systems, or passive design features provided to mix combustible gases within the containment. SPLB reviews the supporting information in the safety analysis report which should include elevation drawings of the containment showing the routing of ductwork and the circulation patterns caused by fans, sprays, or thermal convection. Special attention is paid to interior compartments to ensure that combustible gases cannot collect in them without mixing with the bulk containment atmosphere. SPLB ensures that interior compartments are identified in the safety analysis report and the provisions made to ensure circulation within them are discussed.

   Systems provided to mix the combustible gases within the containment may also be used for containment heat removal, e.g., the fan cooler and spray systems. The acceptability of the design of these systems is considered in the review of the containment heat removal systems in SRP Section 6.2.2.

9. SPLB reviews the manner in which the systems provided to reduce combustible gas concentrations will be operated. The point at which the system is actuated (the control point) will be determined from the safety analysis report. For deliberate ignition systems, the control point is typically core exit temperature exceeding 1200 degrees Fahrenheit.

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

IV. <u>EVALUATION FINDINGS</u>

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

The staff concludes that the design and expected performance of the combustible gas control systems are acceptable and meet the requirements of 10 CFR Part 50, § 50.44, and Criteria 5, 41, 42, and 43. This conclusion is based on the following: [The reviewer should discuss each item of the regulations or related set of regulations as indicated.]

1. The applicant has met the requirements of (cite regulation) with respect to (state limits of review in relation to regulation) by (for each item that is applicable to the review state how it was met and why acceptable with respect to the regulation being discussed):

   a. meeting the regulatory positions in Regulatory Guide(s) ________;

   b. providing and meeting an alternative method to regulatory positions in Regulatory Guide _____, that the staff has reviewed and found to be acceptable;

c. meeting the regulatory position in BTP ___;

d. using calculational methods for (state what was evaluated) that have been previously reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them to be suitably conservative or performed independent calculations to verify acceptability of their analysis; and/or

e. meeting the provisions of (industry standard number and title) that have been reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited above.

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

V. IMPLEMENTATION

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

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

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.

VI. REFERENCES

1. Regulatory Guide 1.7, Revision 3, “Control of Combustible Gas Concentrations in Containment,” dated [DATE OF RULE?].

2. SECY-00-0198, “Status Report on Study of Risk-informed Changes to The Technical Requirements of 10 CFR Part 50 (Option 3) And Recommendations on Risk-informed Changes to 10 CFR 50.44 (Combustible Gas Control),” dated September 14, 2000.

3. SECY-93-087, “Policy, Technical, And Licensing Issues Pertaining to Evolutionary And Advanced Light-water Reactor (ALWR) Designs,” dated April 2, 1993.

4. USNRC, “Station Blackout,” Regulatory Guide 1.155, Revision 0, August 1988.

5. USNRC, “Final Safety Evaluation Report Related to the Certification of the System 80+ Design, Docket No. 52-002,” NUREG-1503, July 1994.

6. USNRC, “Final Safety Evaluation Report Related to the Certification of the Advance Boiling Reactor Design,” NUREG-1462, August 1994.

7. USNRC, “Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design, Docket No. 52-003,” NUREG-1512, September 1998.

8. USNRC, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Regulatory Guide 1.97, Revision 3, May 1983.

9. 10 CFR Part 50, § 50.44, "Combustible Gas Control in Containment."

10. 10 CFR Part 50, § 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Reactors."

11. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems and Components."

12. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

13. 10 CFR Part 50, Appendix A, General Design Criterion 42, "Inspection of Containment Cleanup System."

14. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup System."

15. Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," attached to SRP Section 9.2.5.

16. NUREG/CR-4905, "Detonability of H -Air-Diluent Mixtures," Sandia National Laboratory, June 1987.

17. NUREG/CR-4961, "A Summary of Hydrogen-Air Detonation Experiments," Sandia National Laboratory, June 1987.

18. NUREG/CR-5275, "Flame Facility" (The Effect of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation of Hydrogen-Air Mixtures at Large Scale), Sandia National Laboratory, April 1989.

19. NUREG/CR-5525, "Hydrogen-Air-Diluent Detonation Study of Nuclear Reactor Safety Analyses," Sandia National Laboratory, December 1990.