POLICY ISSUE
NOTATION VOTE

September 14, 2000 SECY-00-0198

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: 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)

PURPOSE:

To provide the second status report on the staff's study of possible risk-informed changes to the technical requirements of 10 CFR Part 50, to provide the staff's recommendations for risk-informed changes to 10 CFR 50.44 ("Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors") that will both enhance safety and reduce unnecessary burden, and to provide policy issues for Commission decision.

BACKGROUND:

In a June 8, 1999, staff requirements memorandum (SRM) on SECY-98-300, the Commission approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50. The Commission specifically directed the staff to pursue the *"study on an aggressive timetable and provide, for Commission approval, a schedule for this activity. The staff should periodically inform the Commission on progress made in the study....if the staff identifies a regulatory requirement which warrants prompt revision..., the Commission should be...provided with a recommended course of action."*

CONTACT:
Mary T. Drouin, RES
415-6675

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 recommended changes resulting 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 plan in a February 3, 2000, SRM. 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, and to develop a communication plan that facilitates greater stakeholder involvement and actively seeks stakeholder participation.

On April 12, 2000, the staff provided its first status report on Phase 1 of this work 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”). This paper provides the staff’s second periodic status report on Phase 1, its recommendations on 10 CFR 50.44, and related policy issues for Commission consideration.

<u>DISCUSSION</u>:

Since the first status report in April 2000, the staff has accomplished a number of activities; it-

- used and revised the framework for studying 10 CFR Part 50.
- identified policy issues for Commission consideration.
- developed recommendations for risk-informed changes to 10 CFR 50.44.
- met with stakeholders (both external and internal) to obtain their input on these activities.
- initiated work to develop risk-informed alternatives to 10 CFR 50.46 (“Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors”) and special treatment requirements.

A summary of each of these activities follows.

Risk-Informed Framework:

The staff has developed a framework that describes the approach, process and guidelines the staff will apply in reviewing, formulating, and recommending risk-informed alternatives to 10 CFR Part 50 technical requirements. An initial version of this framework was attached to SECY-00-0086. The staff is using this framework to develop recommendations for generic changes to the technical requirements and is not applying it on a plant-specific basis. The framework has been tested in risk-informing 10 CFR 50.44 and has also been the subject of comments by stakeholders. It has been updated since the initial version provided in SECY-00-0086 to reflect experience from its use and the comments received; however, it may undergo additional refinement as it is tested against more challenging rules such as 10 CFR 50.46.

The updated framework is provided as Attachment 1 and five of its key features are as follows:

1. The framework utilizes a risk-informed defense-in-depth approach to accomplish the goal of protecting public health and safety. This defense-in-depth approach builds on: (a) the principles in Regulatory Guide 1.174, (b) the Commission's White Paper on risk-informed and performance-based regulation, dated March 11, 1999, (c) the reactor oversight cornerstones, and (d) the Advisory Committee on Reactor Safeguards (ACRS) recommendations on defense-in-depth, as discussed in the ACRS letter to former Chairman Jackson, dated May 19, 1999.

2. The defense-in-depth approach includes elements that are dependent upon risk insights and elements that are employed independent of risk insights. Risk insights are used to set guidelines that-
   - limit the frequency of accident initiating events
   - limit the probability of core damage, given accident initiation
   - limit radionuclide releases during core damage accidents
   - limit public health effects caused by core damage accidents

   Safety function success probabilities (commensurate with accident frequencies, consequences, and uncertainties) are achieved via appropriate
      - redundancy, independence, and diversity,
      - defenses against common-cause failure mechanisms,
      - defenses against human errors, and
      - safety margins.

   The following defense-in-depth elements are employed independent of risk insights:
   - prevention and mitigation are maintained
   - reasonable balance is provided among prevention, containment and consequence mitigation
   - over-reliance is avoided on programmatic activities to compensate for weaknesses in plant design
   - independence of barriers is not degraded
   - the defense-in-depth objectives of the current General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 are maintained

3. The framework considers both design-basis as well as core-melt accidents.

4. The framework considers uncertainties.

5. Consistent with Commission direction in its June 19, 1990, SRM, the staff is using the Safety Goals to define how safe is safe enough. That is, the framework is constructed in such a way that risk-informed alternatives to 10 CFR Part 50 will be developed consistent with this direction (using the subsidiary objectives of the Safety Goals as guidelines). The framework uses quantitative guidelines, based on the Safety Goals and its subsidiary objectives of 10^{-4} per reactor year (ry) for core damage frequency (CDF) and 10^{-5}/ry for large early release frequency (LERF), to assist the staff in determining the appropriate balance between prevention and mitigation and whether or not to recommend a risk-informed alternative to the current requirements.

Policy Issues:

The staff has identified two policy issues for Commission consideration, which are discussed in this section along with a recommended position:

- Selective implementation
- Backfit considerations

Selective Implementation

In SECY-98-300, the staff recommended that implementation of a risk-informed modification be voluntary, but that a licensee should not be allowed to choose which elements of the revised Part 50 to follow. In its response to the staff, the Commission stated that *"risk-informed implementation of Part 50 should be voluntary for licensees. ..... The Commission has disapproved the staff's recommendation that selective implementation not be allowed. This issue is prematurely before the Commission. A future Commission will be better able to judge the issue of selective implementation after the rules are drafted and rulemakings provide comment on this issue as it affects that rule...."*

The staff recognizes that licensees may voluntarily implement a specific risk-informed rule (e.g., 10 CFR 50.44). However, the staff recommends that a licensee not be allowed to select which specific requirements within a risk-informed rule to follow. The risk-informed alternative rules are being developed in an integral fashion and, therefore, represent a balance between reducing unnecessary burden and employing safety enhancements that address risk-significant concerns. Selective implementation of specific requirements within a rule could allow licensees to preferentially reduce burden without also implementing the risk-informed changes that address risk-significant concerns not currently addressed. Such selective implementation is not compatible with the intent of risk-informed regulation, which includes safety improvements justified by risk considerations.

As discussed below, the staff has developed a set of characteristics for a risk-informed version of 10 CFR 50.44. These characteristics reflect an approach of not permitting selective implementation within 10 CFR 50.44. If approved by the Commission, the staff would proceed to use these characteristics to develop a proposed rule and solicit public comment on that rule. As part of this rulemaking, the staff would explicitly request comment on selective implementation. Accordingly, the staff recommends that within the context of development of a risk-informed alternative to 10 CFR 50.44, no selective implementation be allowed.

Backfit Considerations

Risk-informed alternative rules may include a combination of elimination, modification, and addition of requirements. Therefore, the staff does believe that backfit considerations should not be totally ignored since some of the recommended safety enhancements may be sufficiently important to be considered as mandatory changes for all plants. For those risk-informed changes that appear to substantially enhance safety and that have the potential to be cost beneficial, the staff therefore recommends that these changes (i.e., proposed requirements) be sent to the generic issue program for prioritization and consideration as a mandatory change to existing requirements (using provisions of 10 CFR 50.109, "Backfitting"). This will require consideration of alternative means of implementing those changes that enhance safety and conducting detailed cost-benefit analysis.

However, since the licensee may voluntarily implement a risk-informed alternative to a given rule, the staff recommends that a backfit analysis of the risk-informed alternative not be required.

Risk-Informed 50.44:

As discussed in SECY-00-0086, the staff had identified 10 CFR 50.44 as a regulation that "warrants prompt revision" and has developed recommendations for a risk-informed alternative. The current rule was implemented to control combustible gases, such as hydrogen, that could burn or detonate and thereby challenge the integrity of the containment. Consequently, based on knowledge at the time, the following technical requirements were formulated and are contained in 10 CFR 50.44:

- Analytical requirements to address the conditions, source and amount of hydrogen
    1. The type of accident considered, viz. postulated loss-of-coolant accident (LOCA)
    2. The sources of hydrogen (fuel-cladding oxidation, radiolysis, and corrosion)
    3. The hydrogen source term: 5% clad oxidation reaction over a 2-minute period and 75% metal-water reaction of the active clad for Mark III and ice condenser containments[1].

- Physical requirements to mitigate these analytical requirements (first bullet)
    1. Measure hydrogen concentration
    2. Insure a mixed containment atmosphere
    3. Control combustible gas concentration resulting from a postulated LOCA
    4. Inert Mark I and II containments
    5. Install high point vents on the reactor coolant system

[1]10 CFR 50.44 does not impose the 75% metal-water reaction on large dry and subatmospheric containments. However, Generic Safety Issue 121 did address this source term for these containments and found it not to be a challenge.

6. Provide a hydrogen control system (i.e., igniters) for Mark III and ice condenser containments

Other requirements in 10 CFR Part 50 and implementing documents (e.g., regulatory guides) are associated with 10 CFR 50.44. These related requirements and documents have imposed additional “requirements” beyond those stated in 10 CFR 50.44 (e.g., safety-grade continuous monitors for measuring the hydrogen concentration). Therefore, in its evaluation of 10 CFR 50.44, the staff also examined the related regulations and implementing documents.

Based upon current risk information and research results, the staff believes that little to no risk significance or benefit is associated with some of the combustible gas control requirements of this regulation, potentially resulting in unnecessary burden. In addition, the staff also believes that the current requirements do not address some risk-significant concerns from accident scenarios. Therefore, the staff recommends changes to the requirements that represent both a safety enhancement (some of which may have an associated additional burden) and a reduction in unnecessary burden.

Core damage/melt accidents can potentially produce combustible gases (both hydrogen and carbon monoxide) from both fuel-cladding oxidation and core-concrete interaction. Risk insights associated with combustible gas generation and combustion have led to the following conclusions:

- Combustible gases are not a significant challenge to containment integrity for approximately 24 hours after the onset of core damage for:
    - S large dry and subatmospheric containments due to large volume
    - S Mark I and II containments due to inert atmosphere
    - S Mark III and ice condenser containments due to igniters (except for station blackout)

- For station blackout for Mark III and ice condenser containments defense-in-depth is a concern since conditional large early release probabilities from combustible gases can exceed the guideline (0.1) contained in the attached framework document and range from 0.1 to 1.0

- Internal fire and seismic core damage sequences can have the characteristics of station blackout

- Combustible gas concentrations may be a challenge to containment integrity after 24 hours because of:
    - S Core-concrete interactions in large dry, subatmospheric, ice condenser and Mark III containments
    - S Oxygen generation from radiolysis leading to a de-inerted atmosphere in Mark I and II containments

A detailed discussion of the staff’s feasibility study and recommendations is provided in Attachment 2. In summary, the staff considers the work described in Attachment 2 sufficient to establish the feasibility for risk-informed changes to the technical requirements of

10 CFR 50.44 and recommends the following characteristics for a risk-informed alternative to 10 CFR 50.44[2]:

1. Specify in the regulation a specific combustible gas source term using best available calculational methods for a severe accident that includes in-vessel (and ex-vessel) hydrogen and carbon monoxide generation in such a way that the alternative regulation addresses the likely sources of combustible gases. These sources would only address challenges to the containment that could potentially result in a large radionuclide release within 24 hours after the onset of core damage. This is consistent with the approach taken in the staff's review of the Advanced Light Water Reactors (ABWR, System 80+ and AP-600). This recommendation would involve a short-term (~3 months) effort by the staff to perform the calculations and to specify the source term in the regulation that is based on these calculations.

2. Eliminate the requirement to measure hydrogen concentration in containment. Hydrogen monitoring is not needed to initiate or activate the combustible gas control systems for each type of containment, hence hydrogen monitors have a limited significance in mitigating the threat to containment in the early stages of a core-melt accident. Hydrogen monitoring for emergency response purposes is addressed separately from 10 CFR 50.44.

3. Retain the requirement to insure a mixed atmosphere. The intent of this requirement is to maintain those plant design features (e.g., open compartments) that promote atmospheric mixing and is considered an important defense-in-depth element (i.e., meeting the intent of GDC 50).

4. Eliminate the requirement to control combustible gas concentration resulting from a postulated LOCA. This type of accident is not risk significant and the means to control combustible gas concentration (e.g., recombiners) does not provide any benefit for the risk-significant accidents or, if a vent-purge method is used, can result in unnecessary releases of radioactive material to the atmosphere. Long-term combustible gas control is addressed in Item 9 below.

5. Retain the requirement to inert Mark I and Mark II containments. Removal of this requirement would result in the integrity of these containments being highly vulnerable to gas combustion.

6. Retain the requirement for high point vents in the reactor coolant system (RCS). Combustible gases in the RCS can inhibit flow of coolant to the core, therefore, the capability to vent the RCS provides a safety benefit in its ability to terminate core damage.

7. Modify the requirement for the hydrogen control system for Mark III and ice condenser containments to control combustible gas during risk-significant core-melt accidents (e.g.,

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[2]Implementation of this risk-informed alternative would also require changes to other associated regulations and implementing documents.

station blackout). Since the control system uses igniters that are alternating current (ac) dependent, under station blackout conditions, these containments may remain vulnerable to gas combustion. Alternately, if station blackout could be shown by the licensee to be of low enough frequency, with due consideration of uncertainties and defense-in-depth, then the sequence would not be risk significant and the licensee would have complied with the requirement via the current igniter system. Such an approach represents a performance-based aspect of this recommendation.

8. Include a performance-based second alternative within this regulation that would allow a licensee to use risk information and plant-specific analysis on the generation and control of combustible gases to demonstrate that the plant would meet specified performance criteria (e.g., maintain containment integrity for at least 24 hours for all risk-significant events). This may be especially attractive to future plants.

9. Recommend that long-term (more than 24 hours) control of combustible gas be included as part of the licensee's Severe Accident Management Guidelines (SAMG) since combustible gases still pose a challenge to containment integrity in the long term with the possibility of a large, late radionuclide release.

Accordingly, the staff recommends development of a proposed risk-informed alternative to 10 CFR 50.44 consistent with the recommendations in this paper. It is recognized that, since this recommendation is based upon a feasibility study, additional work is required to support the actual rule change. In addition to the calculation of the combustible gas source term discussed earlier, such work would include:

- detailed regulatory analysis on safety enhancements that have the potential to pass the backfit test

- assessing the relation to and need for conforming changes in emergency operating procedures and SAMGs

- assessing the implications of fire and seismic events on the combustible gas control system requirements in Mark III and ice condenser plants

- developing regulatory guides to implement the performance-based aspects of the recommended risk-informed alternative rule.

Also, the rulemaking process will provide opportunities for additional stakeholder feedback on the risk-informed alternative, its technical basis and the additional work needed to support rulemaking. The staff will provide a schedule for this rulemaking 3 months after receiving the SRM on this paper.

These recommendations represent a voluntary risk-informed alternative to the current 10 CFR 50.44, including a performance-based option. In selecting the risk-informed alternative to 10 CFR 50.44, licensees (1) would improve safety by better focusing on the risk-significant challenges from combustible gases, (2) would ensure control of combustible gases during all risk-significant events, and (3) would also eliminate those aspects of the current requirements that provide no safety benefit (e.g., recombiners). As discussed previously, the staff recommends that safety enhancements that have the potential to pass the backfit test be assessed for mandatory application through the generic issue program. The staff estimates that unnecessary burden reduction associated with this alternative is approximately $200K per unit per year (Ref. 2) and that the safety improvement will remove a significant vulnerability (~0.9 conditional containment failure probability) of containment failure during station blackout for Mark III and ice condenser containments. It is recognized that there would be costs associated with the safety improvement; however, the magnitude of these costs is dependent on the means selected by the licensee for implementation. Consistent with the recommendation above on selective implementation, the staff recommends that licensees not be allowed to select individual requirements within the alternative rulemaking (e.g., choose only to eliminate the requirement for measuring hydrogen concentration).

Also, consistent with the policy discussion on backfit considerations in this paper, the staff intends to evaluate the safety issue associated with the Mark III and ice condenser containment igniter power supply as potential backfits through the generic safety issue program.

On November 9, 1999, Mr. Robert Christie of Performance Technology submitted a “request for proposed rulemaking” to the staff on 10 CFR 50.44. As discussed in a January 4, 2000, letter from S. Collins to Mr. Christie, his request has been considered as part of the staff’s study of possible risk-informed changes to 10 CFR 50.44. The recommended risk-informed alternative in this paper addresses Mr. Christie’s request. A comparison of Mr. Christie’s request with the staff’s recommendation is contained in Attachment 3.

Stakeholder Communication:

The staff has held several meetings with stakeholders. These meetings have focused primarily on the framework, and changes to 10 CFR 50.44 and 10 CFR 50.46. The staff also attended an industry workshop on NRC risk-informed activities (in which one session addressed changes to technical requirements in 10 CFR Part 50). In addition, the staff has had several discussions with the ACRS (both the sub- and the full committee) and plans to continue to meet with them on a regular basis. The staff has continued to maintain the interactive Web site[3]. As information is ready for stakeholder review, it is posted to this Web page (and placed in the public document room for those who do not have internet access). Also, stakeholders can provide comments directly to the staff in this Web page; however, as of this date, stakeholders have not exercised this option.

Stakeholder feedback has included:

- Various comments on the framework that questioned whether-
  - the quantitative guidelines are to appear in a regulation.
  - the guidelines are being applied on a generic or plant-specific basis.
  - the Safety Goals are an appropriate measure for the quantitative guidelines.

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[3]The Web site is accessed via the NRC Web site under the Nuclear Reactors icon, and then selecting the “Risk-Informed Part 50 Initiatives" line item, followed by the “Risk-Informed Part 50, Changes to Technical Requirements (Option 3)” line item.

- General agreement that selective implementation of requirements within a regulation should not be allowed.

- Agreement with the staff that the rulemaking on 10 CFR 50.44 needs to be expedited.

- Continue to work closely with the various owner's groups on 10 CFR 50.46.

Future Activities:

The staff has begun work to develop risk-informed alternatives to the current 10 CFR 50.46 and special treatment requirements. The work on 10 CFR 50.46 has involved several public meetings with the Westinghouse Owners Group, which is sponsoring work related to redefining the large break LOCA. RES is planning a public workshop on October 2, 2000, to discuss the latest version of the framework (Attachment 1) and the status and issues associated with risk-informed changes to the technical requirements of 10 CFR 50.46. Making risk-informed changes to the technical requirements of 10 CFR 50.46 has the potential to affect many aspects of plant design and operation. Because of the extent of the potential impacts, we are approaching our study in stages, starting with assessing the possible risk-informed alternatives to the large break LOCA and their implications for requirements related to ECCS performance. Subsequent stages would look at implications for other plant design and performance requirements (e.g., containment) and ECCS acceptance criteria. We expect in December 2000, to be able to report on the first stage and on plans and schedule for the remaining stages. Also, in December 2000, we will report on plans for any other future work, including risk-informed alternatives to the special treatment requirements. This work on risk-informed alternatives to existing special treatment requirements will be coordinated with the ongoing effort on the scope of structures, systems and components subject to these requirements (referred to as Option 2).

RESOURCES:

RES and NRR resources for moving forward, upon Commission approval, with Phase 2 for risk-informing 10 CFR 50.44 are included in the current RES and NRR budgets for FY2001 and FY2002.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objections. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections.

RECOMMENDATION:

The staff recommends that the Commission approve for this paper-

- the proposed staff positions on the two policy issues and
- proceeding with rulemaking (and regulatory analysis) on the risk-informed alternative to 10 CFR 50.44 recommended.

In the interim, the staff will proceed with application of the Option 3 framework in the technical study of additional requirements consistent with its recommendations, unless otherwise directed.

/RA by Frank J. Miraglia Acting For/

William D. Travers
Executive Director
for Operations

Attachments:

1. "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50"

2. “Feasibility Study for a Risk-Informed Alternative to 10 CFR 50.44, ‘Standards for Combustible Gas Control System in Light-water-cooled Power Reactors’”
3. Comparison to R. Christie’s Petition for Rulemaking

References:

1. USNRC, “Safety Goals for the Operation of Nuclear Power Plants; Policy Statement,” *Federal Register,* Vol. 51, p. 30028, August 21, 1986.
2. Letter from J.F. Colvin of Nuclear Energy Institute to Chairman Meserve of the NRC, 1-19-00

Attachment 1

FRAMEWORK FOR RISK-INFORMED CHANGES TO THE TECHNICAL REQUIREMENTS OF 10 CFR 50

Draft, Revision 2

Prepared by
Office of Nuclear Regulatory Research
Division of Risk Analysis and Applications
Probabilistic Risk Analysis Branch

Tom King
Mark Cunningham
Mary Drouin
Alan Kuritzky

Sandia National Laboratories
Allen Camp
Jeff LaChance

ERI Consulting
Eric Haskin

Brookhaven National Laboratory
John Lehner
Vinod Mubayi
Trevor Pratt

August 2000

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1.0 INTRODUCTION

1.1 Background

The NRC's policy statement on probabilistic risk assessment (PRA) encourages greater use of PRA to improve safety decision making and regulatory efficiency (Ref. 1). The NRC has undertaken a number of activities to risk-inform regulations and regulatory processes in order to enhance safety and reduce unnecessary burden.

In SECY-98-300, (Ref. 2) the NRC staff presented the following three options for applying PRA insights to risk-inform existing regulatory requirements in 10 CFR Part 50:

1. Continue ongoing rulemaking activities and risk-informed approaches making no changes to the current Part 50 (Option 1),

2. Change the special treatment rules in Part 50 to modify their scope to be risk-informed, (Option 2), and

3. Make changes to specific requirements in the body of the regulations, including the general design criteria (Option 3).

In a June 8, 1999 Staff Requirements Memorandum (SRM), the Commission approved proceeding with the current rulemakings in Option 1, implementing Option 2, and proceeding with a study of Option 3.

SECY-99-264 (Ref. 3) provides the NRC staff's plan for the study phase of its efforts under Option 3 to risk-inform the technical requirements of 10 CFR 50. The plan consists of two phases:

1. An initial study phase (Phase 1) where recommendations to the Commission on proposed changes will be made, and

2. An implementation phase (Phase 2) where changes recommended in Phase 1 and approved by the Commission will be made.

1.2 Objectives

In SECY-98-300, the staff delineated the following broad objectives for its work to risk-inform 10 CFR Part 50:

- Enhance safety by focusing NRC and licensee resources in areas commensurate with their importance to health and safety,

- Provide NRC with the framework to use to risk information to take action in reactor regulatory matters, and

- Allow use of risk information to provide flexibility in plant operation and design, which can result in burden reduction without compromising safety.

The possible approaches to revising the existing body of regulations under Option 3 include:

- adding provisions to Part 50 allowing the staff to approve risk-informed alternatives to current regulations

- revising specific requirements to reflect risk-informed considerations, and

- deleting unnecessary or ineffective regulations.

The objective of this document is to present a framework that will be used by the NRC staff to guide the development of risk-informed alternative regulations under Option 3. The risk-informed alternatives developed under Option 3 would be voluntary alternatives to current requirements.

1.3 Scope and Limitations

The framework presented herein is a risk-informed defense-in-depth approach, which provides guidance to the NRC staff for its initial efforts to develop risk-informed alternatives to existing regulations (sections of 10 CFR 50) under Option 3. The emphasis is on regulations that impact existing plants. Licensees will have the option to comply with all of the requirements of an existing regulation or with all of the requirements of a risk-informed alternative regulation.

It is anticipated that this framework will continue to evolve as experience is gained in developing risk-informed alternatives. The current guidance is directed toward existing regulations that have an impact on prevention or mitigation of core-damage accidents, because these accidents present the most risk to the public and risk information is most prevalent for such accidents. In the future, the framework can be adapted and extended to apply to regulatory requirements that impact non-core-damage accidents.

The framework is generally consistent with the Regulatory Analysis Guidelines (Ref. 4) and Regulatory Guide 1.174 (Ref. 5). The Regulatory Analysis Guidelines focus on regulatory changes that would decrease risk but impose additional burden. Regulatory changes of this type that are identified under Option 3 and have the potential to pass the backfit rule will be referred to the Generic Safety Issues program to assess the need for mandatory implementation.

Like Regulatory Guide 1.174, the framework also addresses changes that could result in risk increases. Regulatory Guide 1.174, provides guidance to licensees requesting changes to an individual plant's licensing basis. Risk increases associated with such licensee-proposed changes are appropriately evaluated relative to the existing plant risk. An alternative regulation developed under Option 3 will apply to all plants that choose to comply with the alternative rather than the existing regulation. Accordingly, in Option 3 as in the Regulatory Analysis Guidelines, the potential industry-wide risk impact of changes made to comply with alternative risk-informed regulations must be assessed.

Option 2 involves making changes to the overall scope of systems, structures and components (SSCs) covered by those sections of Part 50 requiring special treatment (such as quality assurance, environmental qualification, etc.). Alternative regulations developed in the Option 3 study will reflect the experience gained in Option 2 classification efforts. When possible, approaches that are consistent with Option 2 will be included in risk-informed regulations developed under Option 3.

1.4 Approach

Section 2 describes the risk-informed defense-in-depth approach, which builds on the cornerstones of safe nuclear power operation contained in the Reactor Inspection and Oversight Program. Because the initial focus of the Option 3 efforts is on regulations that impact prevention and mitigation of accidents involving the reactor core, the defense-in-depth strategies are tied to the four reactor safety cornerstones.

Section 3 presents the quantitative guidelines for the framework. These quantitative guidelines will be used by the NRC staff in identifying existing regulations that are candidates for risk-informed change, formulating and evaluating change options, and recommending the changes to be included in alternative, risk-informed regulations.

The quantitative guidelines are not proposed regulatory requirements and will generally not appear in risk-informed regulations; however, they may appear in implementing documents such as regulatory guides when probabilistic analyses are deemed appropriate.

In applying the quantitative guidelines risk increases are only permitted if they are reasonable relative to the Quantitative Health Objectives of the Safety Goal Policy Statement (Ref. 6), and then only they are consistent with the overall defense-in-depth approach. The quantitative guidelines are not proposed regulatory requirements and will generally not appear in risk-informed regulations; however, they may appear in implementing documents such as regulatory guides when probabilistic analyses are deemed appropriate. This reflects an important choice. In theory, one could develop and apply a more generous regulatory framework, one that permits the elimination of all measures not needed for adequate protection (that level of protection of the public health and safety that must be reasonably assured regardless of economic cost). Like the Regulatory Analysis Guidelines and Regulatory Guide 1.174, the framework presented here takes a more restrictive approach. This approach is taken to compensate for PRA limitations and uncertainties, including completeness uncertainty. Safety issues continue to emerge notwithstanding the maturity of the nuclear power industry. Treatment of uncertainties is in Section 4.

Implementation of the framework in the Phase 1 study is described in Section 5. The staff will identify and prioritize candidate regulations for risk-informed changes. If risk information indicates possible holes in existing regulations, these will also be considered. A risk-informed alternative to the technical requirements of a rule will be developed using the framework (as described in Section 5) and recommended to the Commission for approval. This risk-informed alternative will be based on sufficient analysis to show its feasibility. With Commission approval, more detailed regulatory analyses of recommended alternatives will be performed under Phase 2.

2.0 DEVELOPMENT OF THE FRAMEWORK

2.1 Overview

Figure 2-1 illustrates the key elements of the framework. The primary goal is to protect the public health and safety. The framework constitutes a risk-informed, defense-in-depth approach. It will be used by the NRC staff to analyze the effectiveness of existing regulations in supporting the primary goal. When the staff determines that the effectiveness of an existing regulation can be improved, an alternative risk-informed regulation, which is consistent with the framework, is formulated and recommended to the Commission.

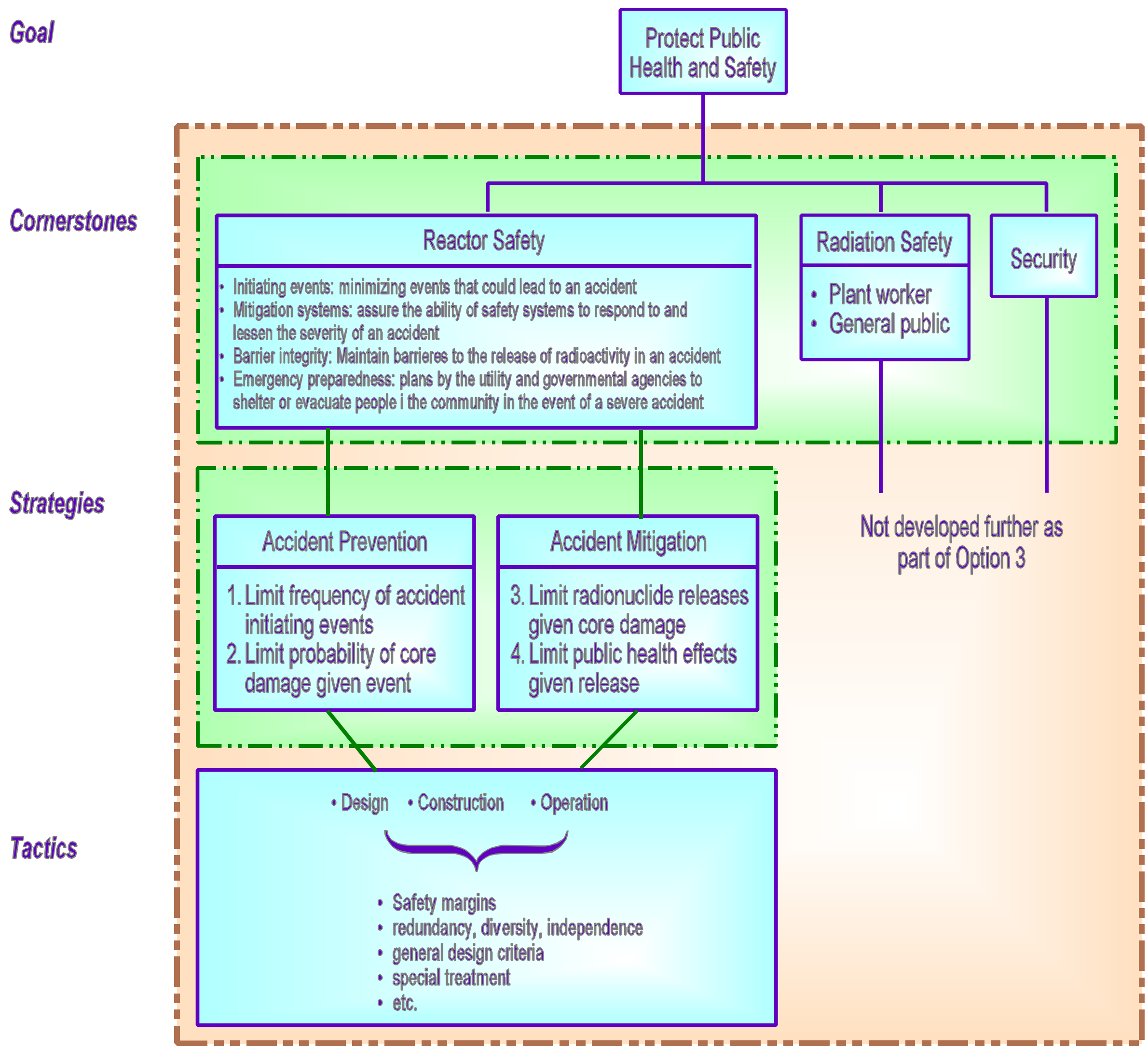

Figure 2-1 Risk-Informed Defense-in-Depth Framework.

The elements of the risk-informed defense-in- depth approach are discussed in Section 2.2. As indicated in Figure 2-1, this approach is consistent with cornerstones of safe nuclear power plant operations, which were identified in the NRC Reactor Inspection and Oversight Program (Ref. 7). Specific strategies and related elements of the framework are used to implement the cornerstones as discussed in the following sections. Quantitative guidelines are developed in Chapter 3.

2.2 Defense-in-Depth Approach

The term defense-in-depth is used to describe applications of multiple measures to prevent or mitigate accidents. The measures can be embodied in SSCs or in procedures (including emergency plans). Defense-in-depth can be applied in various ways. Redundant or diverse means may be used to accomplish a function, the classic example being the use of multiple barriers (fuel, cladding, reactor coolant pressure boundary, spray or scrubbing systems, and containment) to limit the release of core radionuclides. Alternatively, redundant or diverse functional lines of defense may be used to accomplish a goal.

To illustrate, consider the primary goal of protecting the public from nuclear power plant accidents. As indicated in Figure 2-1, the first line of defense is to eliminate initiators that could conceivably lead to core damage. However, it is not possible to eliminate all initiators. The frequency of initiators, although significantly less than before the accident at Three Mile Island Unit 2 (TMI-2), is about 1 per plant year. As a second line of defense, systems such as the Emergency Core Cooling System (ECCS) are provided to prevent core damage should postulated initiators occur. Although such systems are designed for a wide spectrum of initiators and compounding equipment failures, no prevention system is perfect. As a third line of defense, barriers including containment and associated heat and fission product removal systems are required. These barriers would prevent large radionuclide releases for many severe accidents, but scenarios exist in which containment would be breached or bypassed. A fourth line of defense, offsite emergency preparedness, is therefore required.

Defense-in-depth has evolved since the first research reactors were designed in the 1940s. In a recent letter to the NRC Chairman, the Advisory Committee on Reactor Safeguards (ACRS) discusses this evolution, identifies two schools of thought on the scope and nature of defense-in-depth, and recommends an approach for moving forward with risk-informed regulation (Ref. 8),(Ref. 9). The two schools of thought (models) of defense-in-depth are labeled "structuralist" and "rationalist," but they could just as well be labeled "traditionalist" and "risk-based."

The structuralist or traditionalist model asserts that defense-in-depth is embodied in the structure of the regulations and in the design of the facilities built to comply with those regulations. Defense-in-depth requirements are derived by repeated application of the question, "What if this barrier or safety feature fails?" The results of that process are documented in the regulations themselves, specifically in Title 10 of the Code of Federal Regulations.

In contrast, the rationalist (or risk-based) model asserts that defense-in-depth is the aggregate of provisions made to compensate for uncertainty and incompleteness in our knowledge of accident initiation and progression. This is made practical by the ability to quantify risk and estimate uncertainty using PRA methods.

What distinguishes the rationalist model from the structuralist model is the degree to which the rationalist model depends on establishing quantitative safety goals and carrying formal probabilistic analyses, including analyses of uncertainties, as far as the analytical methodology permits. In the rationalist model, the exercise of engineering judgement, to determine the kind and extent of defense-in-depth measures, occurs after the capabilities of the analyses have been exhausted.

The approach adopted herein recognizes the relevance of both structuralist and risk-based considerations. From a structuralist viewpoint, the approach requires accident prevention and mitigation strategies and supporting elements. Reflecting the rationalist view, probabilistic insights are used in implementing the required strategies and elements. The approach used in Option 3 is summarized in the following working definition:

Defense-in-depth is the approach taken to protect the public by applying the following strategies in a risk-informed manner:

1. *limit the frequency of accident initiating events*
2. *limit the probability of core damage given accident initiation*
3. *limit radionuclide releases during core damage accidents*
4. *limit public health effects due to core damage accident*

The strategies consider the following defense-in-depth elements:

- *reasonable balance is provided among the strategies (as shown in Figure 3-1).*
- *over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.*
- *independence of barriers is not degraded.*
- *safety function success probabilities commensurate with accident frequencies, consequences, and uncertainties are achieved via appropriate*
  - *redundancy, independence, and diversity,*
  - *defenses against common cause failure mechanisms,*
  - *defenses against human errors, and*
  - *safety margins*
- *the defense-in-depth objectives of the current General Design Criteria (GDCs) in Appendix A to 10 CFR 50 are maintained.*

The four strategies emphasizes defense against core damage accidents, which dominate the risk to public health and safety posed by existing plants. Quantitative guidelines are developed in Chapter 3 to characterize a reasonable balance among the preventive and mitigative strategies. For risk significant accidents in which one or more of the four strategies are precluded (e.g., containment bypass accidents), the remaining strategies may be more tightly regulated; that is, regulations should provide a very high confidence in the remaining strategies. Similarly, more stringent requirements may be imposed in the presence of large uncertainties regarding the effectiveness of one of the strategies.

The supporting elements specifically listed in the working definition have, with the addition of safety margin, been adapted from the defense-in-depth elements listed in Regulatory Guide 1.174. The importance of the supporting elements in the presence of uncertainties, in particular the use of safety margin, is discussed in Section 4.

As indicated by the final element of the working definition, effective practices are preserved. Emergency planning will be maintained to support the fourth strategy. Requirements that fuel design limits not be exceeded in anticipated operational occurrences (AOOs) and that the extent of fuel damage be limited in design basis accidents (DBAs) will be maintained. Preserving an effective practice does not preclude developing risk-informed changes to the practice. For example, risk insights will likely be used to identify alternative, risk-informed DBAs to be analyzed. Similarly, risk-informed changes to GDCs are not precluded. For example, it has been suggested that a number of requirements related to fuel design limits during normal operation could be eliminated because their intent is being met for commercial reasons, and the requirements are not risk significant. Also, the risk significance of failure events prescribed for DBAs in the GDCs will be evaluated based on PRA insights.

2.3 Cornerstones and Strategies

In the process of developing risk-informed improvements to the NRC Reactor Inspection and Oversight Program (Ref. 10), general agreement was reached with the nuclear industry and the public regarding the following cornerstones of safe nuclear power plant operations:

Reactor Safety Cornerstones

1. Initiating Events - Minimizing events that could lead to an accident

2. Mitigation Systems - Assure the ability of safety systems to respond to and lessen the severity of an accident

3. Barrier Integrity - Maintain barriers to the release of radioactivity in an accident

4. Emergency Preparedness - Plans by the utility and governmental agencies to shelter or evacuate people in the community in the event of a severe accident

Radiation Safety Cornerstones

5. Plant Worker - Minimize exposure during routine operations

6. General Public - Provide adequate protection during routine operations

Security Cornerstone

7. Physical protection of plant and nuclear fuel

The four reactor safety cornerstones are directly addressed in PRAs and are, therefore, most relevant to the initial Option 3 efforts. As illustrated in Figure 2-1, the four reactor safety cornerstones are reflected in the framework by the four defense-in-depth strategies. The strategies seek both to prevent core damage accidents and to mitigate the public impact should a core damage accident occur. The two preventive strategies are:

- limit the frequency of accident initiating events (initiators), and

- limit the probability of core damage given accident initiation.

The two mitigative strategies are:

- limit radionuclide releases during core damage accidents, and

- limit public health effects due to core damage accidents.

Except for the implied emphasis on core damage accidents, Strategy 1 is identical to Reactor Safety Cornerstone 1. Similarly, for core damage accidents, Strategy 4 is equivalent to Reactor Safety Cornerstone 4, and Strategies 2 and 3 are functionally equivalent to Reactor Safety Cornerstones 2 and 3.

The four defense-in-depth strategies are intentionally more focused than the reactor safety cornerstones. The cornerstones also apply to accidents that can not lead to core damage (for example fuel-handling, fuel-storage, and radwaste storage tank rupture accidents). The strategy statements may in the future be modified to address non-core-damage accidents; however, emphasis on core damage accidents is appropriate for the initial efforts to risk-inform existing regulatory requirements.

The radiation safety and security cornerstones are part of the overall approach, but generally secondary considerations in making risk-informed changes to the existing regulatory requirements. This is because they are not well-treated in probabilistic risk assessments.

In describing the cornerstones and strategies, the words “limit,” “prevent,” and “contain” are relative rather than absolute. Cutting a failure rate in half "prevents" half the failures that would otherwise occur in a given time period, and some fixes last for the life of a plant. However, it is not possible to prevent all accident initiators or to eliminate the possibility of core damage or containment failure for all conceivable accidents. All four strategies are applied to compensate for the limitations of the individual strategies; issues related to PRA scope, level of detail, and technical adequacy; and uncertainty, in particular completeness uncertainty.

2.4 Other Framework Elements

As indicated in Figure 2-1, other elements are applied to support the cornerstones and related strategies. These elements are referred to as tactics to distinguish them from the four defense-in-depth strategies. Existing regulatory requirements apply a wide variety of tactics. Some tactics such as quality assurance are broadly applicable to all four strategies. Other tactics, are used to address a particular type of concern. Safety margin is often applied to provide a high degree of confidence that a design or process will provide a needed function. (Safety margin is discussed further in Section 4.) Other tactics may only be applicable to specific strategies or accident types. No attempt is made to present a comprehensive list of tactics. Assessing which, if any, tactics are required to support a given regulation is part of the Option 3 study. The primary responsibility for implementing tactics, whether required by regulations or not, resides with the licensee.

The single failure criterion is a tactic that is to be examined in the Option 3 study. Specifically, "the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system" have yet to be developed (10 CFR 50 Appendix A). Insights from probabilistic risk assessments regarding the risk significance of passive single failures in fluid systems will be reviewed, and options for resolving this issue will be delineated consistent with the quantitative guidelines developed in Section 3.

3.0 QUANTITATIVE GUIDELINES FOR THE FRAMEWORK

Quantitative guidelines for the preventive and mitigative defense-in-depth strategies are developed in this section. These guidelines are applied by the NRC staff to assess the effectiveness of existing regulations, to formulate and compare risk-informed options to existing regulatory requirements, and to develop risk-informed alternative regulations.

In the context of integrated decisionmaking, the acceptance guidelines should not be interpreted as being overly prescriptive. The quantitative guidelines are not proposed regulatory requirements. They reflect a desired level of safety against which to compare industry-averaged risk measures; a level that is "safe enough" based on the Commission's Safety Goal Policy Statement while providing reasonable balance among the defense-in-depth strategies.

As a starting point for developing quantitative guidelines, consider the Quantitative Health Objectives (QHOs), which were originally set to as a measure of "safe enough":

- "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accident to which members of the U.S. population are generally exposed."

- "The risk to the population in the area of nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes."

These QHOs have been translated into two numerical objectives, as follows:

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accident, etc., is about $5x10^{-4}$ per year. One-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than $5x10^{-7}$ per reactor year (ry). The "vicinity" of a nuclear power plant is understood to be a distance extending to 1 mile from the plant site boundary. The "average" individual risk is determined by dividing the number of prompt or early fatalities (societal risk) to 1 mile due to all accidents, weighted by the frequency of each accident, by the total population to 1 mile and summing over all accidents.

- "The sum of cancer fatality risks resulting from all other causes" is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or $2x10^{-3}$ per year. One-tenth of one percent of this implies that the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to $2x10^{-6}$/ry. The "area" is understood to be an annulus of 10-mile radius from the plant site boundary. The cancer risk is also determined on the basis of an "average individual," i.e., by evaluating the number of latent cancers (societal risk) due to all accidents to a distance of 10 miles from the plant site boundary, weighted by the frequency of the accident, dividing the total population to 10 miles, and summing over all accidents.

Unfortunately, the QHOs are difficult to apply in making risk-informed changes to the existing regulations. PRAs often do not proceed to Level 3, that is, to the quantification of public health risks and even if they did, their calculation is dependent upon many factor outside the licensee's control (e.g., population density).

In addition, simply replacing existing existing regulations with the QHOs would not be risk-informed. It would not assure reasonably balanced defense-in-depth approach. To illustrate, consider the following example. Even at a densely populated U.S. site, if a plant's core damage frequency is 10^{-4} per year or less, the latent cancer QHO is generally met with no credit taken for containment. The early fatality QHO is more restrictive than the latent cancer QHO. If a plant's large early release frequency is 10^{-5}/yr or less, the early fatality QHO is generally met. Conceivably, both QHOs could be met by reducing a plant's CDF to 10^{-5}/yr or less with no containment and no preplanned offsite protection actions. This would not constitute a risk-informed approach.

What is required for a risk-informed approach are quantitative measures and guidelines that can be used to describe and indicate the effectiveness of the defense-in-depth strategies. The measures and guidelines proposed for this purpose are summarized in Figure 3-1. They are generally consistent with those in current use (e.g., (Ref. 11)(Ref. 12)).

Figure 3-1 Quantitative Guidelines for Risk-Informed Changes to Regulatory Requirements.

	Accident Prevention		Accident Mitigation	
	Core Damage Frequency $\leq 10^{-4}$/year		Conditional Large Early Release Probability $\leq 10^{-1}$ (Note 5)	
	Limit frequency of accident initiating events	Limit probability of core damage given event	Limit radionuclide releases given core damage	Limit public health effects given release
	Initiator Frequency	Conditional core damage probability	Conditional large early release probability	Conditional individual fatality probability
Frequent initiators	≥ 1/year	$\leq 10^{-4}$	$\leq 10^{-1}$	Note 3
Infrequent initiators	$\leq 10^{-2}$/year	$\leq 10^{-2}$	$\leq 10^{-1}$	Note 3
Rare initiators	$\leq 10^{-5}$/year	Note 4	Note 4	Note 3

Notes:
1. The product across each row gives a large early release frequency of $<10^{-5}$/year.
2. It is preferable that no single type of initiator cause a large fraction of any frequency guideline.
3. No quantitative guideline is proposed for the fourth strategy, the LERF guideline is used as a surrogate.
4. For rare initiators, emphasis is placed on Strategy 1, limit initiator frequency.
5. Measures to mitigate late large releases are also appropriate. A conditional probability of a late large release (up to 24 hours after the onset of core damage) of $\leq 10^{-1}$ is proposed.

Two methods of quantitatively assessing the level of protection against accidents at a given nuclear power plant are also depicted in Figure 3-1:

- a prevention-mitigation assessment considers the strategies in pairs,
- an initiator-defense assessment considers the strategies individually.

The quantitative guidelines are discussed in the context of these two assessment methods in the following sections. In this context, mean risk measures quantified in full-scope, plant-specific PRAs would ideally be compared to the quantitative guidelines. Full scope PRAs address internal and external initiating events as well as accidents initiated in all operating modes. The frequencies in Figure 3-1 are, accordingly, stated per calender year rather than per year of reactor operation. Other relevant considerations regarding the terms core damage frequency (CDF), large early release frequency (LERF), and large late release are discussed in Section 3.4. Practical considerations regarding the application of the quantitative guidelines to the Option 3 study in the presence of uncertainties and plant-to-plant variations is provided in Section 4.

3.1 Prevention-Mitigation Assessment, Consider the Four Strategies in Pairs

As indicated in Figure 3-1, a prevention-mitigation assessment examines the effectiveness of the strategies in pairs.

To assess the effectiveness of the two preventive strategies, a plant's mean CDF is compared to the quantitative guideline of 10^{-4} per year. If the CDF is 10^{-4} per year or less the latent-cancer QHO is generally met.

To assess the effectiveness of the two mitigative strategies the conditional probability of a large early release given a core damage accident is compared to the guideline of 10^{-1}. (The term large early release is explained in Section 3.4.) The LERF is the product of CDF and the conditional probability of a large early release given core damage. Therefore, if the CDF and conditional probability of large early release guidelines are both met, LERF will be 10^{-5} per year or less. Based on Level 3 PRA results, the early-fatality QHO is generally met if LERF is 10^{-5} per year or less.

The use of a LERF guideline developed from the early-fatality QHO, does not imply that risks associated with late containment failures can or will be ignored. Measures to remove heat from containment and to reduce the concentrations of radionuclides that could otherwise result in later large releases are also appropriate to provide defense against situations in which evacuation is precluded or rendered ineffective, to protect plant workers, and to help ensure plant radiological conditions allow implementation of severe accident management guidelines. The LERF guideline does not adequately address this situation and thus an additional guideline applicable out to approximately 24 hours is proposed to assess the performance of containment and containment engineered safety features. Specifically, a guideline of 10^{-1} or less is applied to the conditional probability of a large late release (i.e., one that does not contribute to LERF, but occurs within approximately 24 hours of the onset of core damage). The potential for late large releases is discussed further in Section 3.4.

Based on existing PRAs the proposed quantitative guidelines provide a reasonable balance between the preventive and mitigative strategies. Uncertainties tend to grow as postulated accidents proceed in time, and existing containments were not designed for severe accidents, A more stringent guideline for the conditional probability of a large early release given a core damage accident could, therefore, be impractical for many plants. On the other hand setting the guideline for CDF at 10^{-4} per year emphasizes the preventive strategies where PRA results are most plentiful and accurate.

3.2 Initiator-Defense Assessment, Consider the Four Strategies Individually

In an initiator-defense assessment events that could conceivably initiate a core damage accident are divided into three categories: anticipated, infrequent, and rare. For each initiator category a quantitative guideline is established for each of the four defense-in-depth strategies. Accident sequences postulated during low power should be weighted according to the anticipated duration of the shutdown period. For example, an accident that can only happen during one week every two years but which has an occurrence probability of 10^{-4} during that week has a frequency of $(10^{-4}/\text{week})*(1 \text{ week}/2 \text{ years}) = 5\times10^{-5}/\text{year}$.

In PRAs, accidents are binned (grouped) by their initiators. Accidents that cause similar behavior and require functionally identical responses to avoid core damage or containment failure are binned together. For example, loss-of-coolant accidents (LOCAs) are often classified as small, intermediate, or large depending on the systems required to respond. Some accidents types (e.g., anticipated transients without scram [ATWS] and station blackout [SBO]) reflect functionally similar sequences of events. For Option 3, three groups of initiator are defined as frequent, infrequent, and rare initiator categories. Each of these are described below.

Anticipated initiators are either expected to occur or may well occur during the life of an individual plant. Examples include inadvertent opening of a steam generator relief or safety valve, steam pressure regulator malfunction, reactor coolant pump trip, and loss of offsite power. The term anticipated operational occurrence (AOO), as used in safety analysis reports, describes a sequence of events started by an anticipated initiator and compounded by one or more single active failures. Plants are generally designed to withstand anticipated operational occurrences with no reactor coolant system or containment damage.

The frequency of a significant group (bin) of anticipated initiators is typically greater than 10^{-2} per year. Anticipated initiators may be risk-significant if multiple failures of responding systems and components lead to core damage. Since the 1979 accident at TMI-2, industry efforts to reduce the frequency of anticipated initiators have been quite successful. Licensees are motivated to reduce the frequency of anticipated initiators by economic as well as safety considerations, and their performance is easily monitored. Therefore, no quantitative guideline for the frequency of anticipated initiators is required to risk-inform existing regulatory requirements. Figure 3-1 simply indicates that the frequency of such initiators is typically on the order of one per year.

The quantitative guideline proposed for the probability of core damage conditional on the occurrence of an anticipated initiator is 10^{-4}. This is consistent with previous Commission Guidance which approved the use of a 10^{-4} CDF objective.

A quantitative guideline of 10^{-1} or less is set for the conditional probability of a large early release given an anticipated initiator that leads to core damage. A quantitative guideline of 10^{-1} or less is also set for the conditional probability of a large late release (i.e., one that does not contribute to LERF, but occurs within approximately 24 hours of the onset of core damage). These are the same guidelines used in the prevention-mitigation assessment. Under the proposed defense-in-depth approach, the fact that core damage results from an anticipated initiator is irrelevant to the level of containment performance desired given core damage. The combination of 10^{-4} CDF and 10^{-1} for conditional probability of a large early release will help ensure the LERF objective of 10^{-5}/ry.

A quantitative guideline has not been set for the fourth line of defense, that is, for the probability of acute fatality given a large early release. This risk measure has not been explicitly considered in past studies, but NUREG-1150 and other Level 3 risk assessments demonstrate that the QHOs are generally met if the quantitative guidelines for the first three strategies are met. In part, this is because wind and rain patterns generally assist in limiting the fraction of the population exposed to offsite radionuclide releases. Offsite protective actions are, nevertheless, an essential element of the risk-informed defense-in-depth approach.

Infrequent initiators are not expected to occur over the life of any single plant but may, nevertheless, occur in the population of plants and could be risk significant. The frequency of a significant group (bin) of infrequent initiators is typically less than 10^{-3} per year. Existing plants were designed to withstand many infrequent initiators including pipe breaks in nuclear steam supply systems (NSSSs) and safe-shutdown earthquakes.

The quantitative guideline is less than10^{-2}/ry for the frequency of all initiators in the infrequent category. On an industry-wide basis it is possible to monitor performance against this quantitative guideline. The quantitative guideline for the conditional probability of core damage given an infrequent initiator is 10^{-2} to ensure a CDF less than 10^{-4}. Based on existing PRAs the proposed quantitative guidelines provide a reasonable balance between initiator prevention and core damage prevention. The guidelines for the two mitigative strategies are again a conditional probability of a large early release of 10^{-1} or less and a conditional probability of a large late release of 10^{-1} or less.

For accidents in which one or more of the four high-level defense-in-depth strategies is precluded, the individual strategy guidelines may be less important than their products; that is, more emphasis needs to be placed on the strategies that remain. For example, consider a PWR interfacing-system loss-of-coolant accident (ISLOCA) in which containment is bypassed. The early containment failure probability is 1.0, therefore the quantitative guideline of 10^{-1} cannot be achieved. Since no special ECCS is provided for ISLOCAs, there is a need to limit the relative frequency of such LOCAs and consider them in emergency planning.

Rare initiators are those excluded from the anticipated and infrequent categories because they are extremely unlikely. Examples of rare initiators include aircraft impact, meteor strikes, and very large earthquakes. As a quantitative guideline, the total frequency of all rare initiators should be 10^{-5} per year or less. Although some rare initiators could fail containment or preclude emergency response, this is not true for all rare initiators, and existing Level 3 PRAs indicate the rare initiator frequency goal of 10^{-5}/yr should not cause the QHOs to be exceeded.

There should be a high level of confidence that the collective frequency of all rare initiatiors is less than 10^{-5} per year. The complete set of rare events cannot be delineated with certainty, and uncertainties in the frequencies of rare events are generally large. Initiators of a specific type (bin) should, therefore, be classified as infrequent only if their frequency is demonstrably less than 10^{-6} per year. Current regulatory guidance imposes even more stingent frequency criteria in screening for external initiators to be addressed in safety analysis reports (Ref. SRP 2.3.3).

The risk-informed defense-in-depth approach does not ignore rare events. Tactics such as research, inspection, testing, and monitoring are applied to validate the low frequencies of rare initiators. Generally, however, a risk-informed regulation will not require plant structures, systems, and components be specifically designed to cope with rare initiators. Existing plant features provide some degree of protection against core damage and radionuclide releases for many rare initiators, and risks posed by rare initiators should certainly be addressed in PRAs. However, to focus on reducing risks associated with rare initiators would draw attention away from, and potentially increase risks associated with, more likely initiators.

3.3 Additional Thoughts on Quantitative Guidelines

When the first two strategies, prevent initiators and prevent core damage, are considered as a pair, the relevant quantitative guideline is a CDF less than 10^{-4} per year. When these strategies are considered individually, the products of the quantitative guidelines for the two strategies is the 10^{-4} per year CDF quantitative guideline. That is, meeting the risk-informed regulations should be consistent with achieving a CDF of less than 10^{-4} per year. To meet such a guideline, the regulations should assure a higher response reliability (perhaps more redundancy and diversity) for more frequent initiators.

A different approach has been taken for rare events. Some of these events, should they occur, have the potential to progress directly to offsite releases of radionuclides. Because the core damage prevention and containment strategies may be unavailable for rare initiators, the frequency quantitative guideline for rare initiators is set more stringently than 10^{-4} per year. Specifically, the quantitative guideline is less than 10^{-5} rare initiators per year with no single type of rare initiator being allowed to account for the entire guideline.

The fourth high-level defense-in-depth strategy involves emergency planning and response, which are essential for protecting the public health and safety. Although a quantitative guideline has not been set for this strategy, credit has been taken for its effectiveness in establishing subsidiary quantitative guidelines compatible with the QHOs for the first three strategies. As noted earlier, pre-planned protective actions may be particularly important for accident scenarios in which one or more of the first three strategies are compromised. For example, for an ISLOCA, which bypasses containment, an early containment failure guideline cannot be used; therefore, the fourth strategy becomes necessary.

The product of the quantitative guidelines for the two strategies in method (1) and the three strategies for each of the three initiator types in method (2) is a LERF of $<10^{-5}$ per year. As stated earlier, this generally assures that the early fatality QHO of #5x10^{-7} per year will be met. Setting the individual strategy quantitative guidelines to yield a lower aggregate value would be unnecessarily conservative.

3.4 Core Damage and Large Release

Many of the risk measures and quantitative guidelines in Figure 3-1 are frequencies or conditional probabilities of core damage or large early release. It is, therefore, appropriate to consider these terms further.

To be risk significant, core damage must involve the release of fission products from the fuel. A risk-significant level of core damage exceeds that specified in the ECCS acceptance criteria of 10 CFR 50.46. The ECCS acceptance criteria permit only one percent of the cladding to be oxidized. Only a fuel-clad gap release would occur given this level of damage. The purpose of the ECCS acceptance criteria is, however, not to establish a risk-significant level of core damage but to set a level of core damage appropriate for a design basis accident.

A typical PRA criteria for core damage requires the water level to be below a certain level with no imminent restoration of coolant to the core region so a melt release of fission products from the fuel is assured. This corresponds roughly to the point where computer analyses become complicated by geometry changes associated with melting and relocation of core materials.

In Regulatory Guide 1.174, LERF is described as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines.

Not every containment bypass or early failure would result in a large release. To be risk-significant containment leakage must far exceed the design basis containment leak rate. Containment failure modes that result in scrubbed releases or leak paths that are isolated before the onset of significant core damage generally do not lead to large releases. However, it needs to be recognized that the determination of what constitutes a scrubbed release is dependent upon several factors, including the depth of the water pool and pool temperature.

In many postulated severe accidents, reactor vessel bottom head failure occurs before effective evacuation. Substantial containment loads accompany bottom head failure. Large releases, therefore, tend to be most likely before or shortly after vessel bottom head failure. Containment failure resulting in a large early release is less likely for degraded-core accidents in which core degradation is arrested in time to prevent vessel bottom head failure.

For some plants, large releases could occur hours after reactor vessel bottom head failure. An example would be a release due to containment overpressurization or high temperature while core-concrete interactions are proceeding in the absence of an overlying water pool. Containment heat removal systems may be inoperable in this scenario, and natural processes would take hours after the completion of core-concrete interactions to remove radionuclides from the containment atmosphere.

Effective evacuation can mitigate the threat of acute health effects offsite given such a delayed large release. However, there are accidents in which external events may preclude or hinder evacuation efforts. Plant workers would also need to be protected from any delayed large release. As indicated in Section 3.1, a quantitative guideline has also been included to reflect the need for defense-in-depth against the threats posed by such delayed releases. Specifically, the conditional probability of a large late release should be 10^{-1} or less. Late in this context extends to approximately 24 hours after the onset of core damage. This period is generally sufficient to provide for significant reduction of airborne radionuclide concentrations in containment. The use of a 24-hour time period forces the staff to review the effectiveness of containment and containment engineered safety features beyond vessel breach. It also represents a reasonable delay for interventions (e.g., controlled elevated containment venting) to cope with long-term or gradual energy releases to containment.

4.0 TREATMENT OF UNCERTAINTIES

In making risk-informed changes to the existing regulatory requirements it is important to consider the treatment of uncertainties from two perspectives: (1) assessing the impact of contemplated changes relative to the quantitative guidelines and (2) developing risk-informed options to existing requirements that reduce the potential impact of uncertainties on the decisionmaking process. Both perspectives are discussed in this section.

4.1 Developing Risk-Informed Alternative

To the extent possible, a risk-informed alternative to existing technical requirements of a regulation will be delineated in such a way that the impact of uncertainties on the decisionmaking process is accounted for.

Regulatory requirements impacting the design of existing plants were, for the most part, promulgated before PRA was broadly applied. Yet, it is fair to say that a driving intent of existing regulations is to define the design envelope of plants such that events within the design envelope are not significant contributors to risk. PRAs and IPEs tend to confirm that this intent has been realized; that is, risk-dominant accident scenarios are generally those involving initiators or multiple failures not postulated in the design of existing plants.

Risk-informed regulations will continue to assure that events within the design envelope are not significant contributors to risk. For example, for routine operation, including anticipated operational occurrences, requirements necessary to minimize cladding failures will be retained, and risk significant levels of core damage will not be accepted for design basis accidents.

In considering a change to an existing regulatory requirement it is important to estimate the overall impact on risk measures of the actual plant changes (to SSCs, inspections, testing, operating procedures, training, emergency plans, etc.) that would ensue. An overall assessment is required to preclude unintended repercussions. For example, if it were demonstrated that very large pipe breaks could be excluded from consideration under the Emergency Core Cooling Systems (ECCS) acceptance criteria of 10 CFR 50.46, such breaks might still represent reasonable design-basis events for containment to account for uncertainties.

The alternative promulgated may be impacted by the type of uncertainty that exists. Although the quality and coverage of risk assessments continues to evolve, completeness uncertainty can never fully be eliminated. Completeness uncertainty associated with the scope of a reference PRA should be addressed by applying risk insights from other relevant PRAs. Completeness uncertainty associated with what has not been thought of or cannot currently be modeled is a principal reason for adopting the high-level defense-in-depth approach and strategies described in Sections 2.2 and 2.3.

Safety margin is often appropriate to compensate for model uncertainty regarding the loads and capacities, for example, to keep passive failures of mechanical components from dominating the failure rates of responding systems. The use of safety margin is discussed further in the next subsection

4.2 Safety Margin

The treatment of uncertainty from the design basis perspective involves the notion of safety margin. Colloquially, terms like safety margin and safety factor imply a measure of the conservatism employed in a design or process to assure a high degree of confidence that it will work to perform a needed function.

There are, in the literature, many different definitions of safety margin. Some are probabilistic. Others are deterministic. For example, safety margin is sometimes defined as the ratio of the ultimate failure stress to the design stress. In delineating risk-informed options to existing regulatory requirements, probabilistic considerations will be applied to the extent possible. The following is typical of a probabilistic definition (Ref. 13): safety margin is the probability (or level of confidence) that a design or process will perform an intended function.

To illustrate the significance of a probabilistic approach, consider the common question: Will the capacity of a structure, system, or component (SSC) be exceeded during an accident? If there is no uncertainty in the imposed stress and no uncertainty in the capacity of the SSC, there is no uncertainty in the answer. Assume a known stress is only slightly less than a known capacity. Replacing the SSC with one that is twice as strong would be useless because the failure probability would still be zero.

Generally, of course, there is uncertainty in the imposed stress, the capacity, or both, and the greater the uncertainties, the greater the need for safety margin. Safety margin may indicate the probability that an uncertain stress exceeds a known capacity or the probability that a known stress exceeds an uncertain capacity. Often there is uncertainty in both the stress imposed and the capacity. In some of these cases, the overlap of the stress and performance distributions can be quantified. More frequently, in formulating regulatory requirements, acceptance criteria or failure criteria are delineated to, in effect, fix the capacity so that safety margin can be stated as the probability of exceeding the acceptance criteria. For example, compliance with the ECCS acceptance criteria of 10 CFR 50.46 can be demonstrated using best-estimate codes provided that “uncertainty is accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria, there is a high level of probability that the criteria would not be exceeded.”

The working definition of safety margin does not preclude the use of conservative or bounding calculations to demonstrate acceptable safety margin. For example, ECCS calculations based on 10 CFR 50 Appendix K, provide a conservative alternative to best-estimate calculations with uncertainty propagation. However, consistent with the intent to use probabilistic considerations where possible, safety margin could be applied to assure a component's structural failure probability is comparable to the probabilities of other failure modes. There is little to be gained by requiring more capacity as long as the structural failure cannot cause other failure events.

Excessive safety margins benefit neither the NRC nor the nuclear industry. Excessively conservative requirements can, in fact, lead to incorrect safety conclusions and regulatory decisions, that may actually reduce plant safety by masking issues of higher safety significance. Mandated excessive conservatism can also produce artificial regulatory concerns.

What constitutes adequate margin and what constitutes excess margin? The answer to this question will always involve engineering judgement. Preliminary guidance for the Option 3 study is offered below, but it is anticipated that guidance regarding safety margin will evolve as the study progresses.

Safety margin is imposed to account for uncertainties in data and models by conservatisms placed in acceptance criteria and methods for demonstrating compliance with acceptance criteria. The approach preferred for the Option 3 study is (1) to specify reasonable safety margin in acceptance criteria based on probabilistic considerations and risk insights, and (2) to use best-estimate code calculations with uncertainty propagation to demonstrate compliance based on a computed 95th percentile. When this approach is precluded, an attempt will be made to achieve an equivalent level of safety margin in order to avoid excessive conservatism.

4.3 Types of Uncertainty

Aleatory uncertainty is that addressed when the events or phenomenon being modeled are characterized as occurring in a "random" or "stochastic" manner, and probabilistic models are adopted to describe their occurrences. This aspect of uncertainty gives PRA the probabilistic part of its name.

Epistemic or state-of-knowledge uncertainty is that associated with the analyst's confidence in the predictions of the PRA model. It reflects the analyst's assessment of how well the PRA model represents the actual system being modeled. As such, it generally varies from analyst to analyst.

Aleatory uncertainty is built into the structure the PRA model. Uncertainty in the results obtained from the PRA model is epistemic. Epistemic (state-of-knowledge) uncertainties are commonly divided into three classes: parameter uncertainty, model uncertainty, and completeness uncertainty.

Parameter uncertainties are those associated with the values of parameters of the PRA models. They are typically characterized by establishing probability distributions on the parameter values. These distributions can be interpreted as expressing the analyst's degree of belief in the values these parameters could take, based on his state of knowledge and conditional on the underlying model being correct. It is reasonably straightforward to propagate the distribution representing uncertainty on the basic parameter values to obtain probability distributions on Level 1 PRA results such as core damage frequency and accident sequence frequencies. Uncertainty characterization is much more difficult in Level 2 PRAs, and generally impractical in Level 3 PRAs.

Model uncertainties are those associated with incomplete knowledge regarding how models used in PRAs should be formulated. Such uncertainties arise, for example, in modeling human performance; common cause failures; and mechanistic failures of structures, systems and components; and large-early releases. Model uncertainties grow in number and magnitude as one proceeds from Level 1 to Level 2 and 3 PRAs.

In some cases, where well-formulated alternative models exist, PRAs have addressed model uncertainty by using discrete distributions over the alternative models, with the probability (or weight) associated with a specific model representing the analyst's degree of belief that the model is the most appropriate. For example, different hypotheses lead to different seismic hazard curves. Discrete weights summing to one are assigned to these curves. Another approach to addressing model uncertainty is to adjust the results of a single model through the use of an adjustment factor. Using such approaches, model uncertainty can be propagated through the analysis in the same way as parameter uncertainty.

More typically, however, the use of different models would result in the need for a different structure (e.g., with different thermal hydraulic models used to determine success criteria). In such cases, although the uncertainties are recognized, they are not quantified. Assumptions are made and specific models are adopted. Unquantified model uncertainty also arises because PRAs bin the continuum of possible plant states in a discrete way. Such approximations introduce biases (model uncertainties) into the results.

In interpreting the results of a PRA, it is important to develop an understanding of the impact of a specific assumption or choice of model on the predictions of the PRA. This is true even when the model uncertainty is treated probabilistically, since the probabilities, or weights, given to different models are subjective. The impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies, or they may be addressed using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models. The impact of making specific modeling approximations may be explored in a similar manner.

Completeness uncertainty refers to things that are not modeled in a PRA. This includes risk contributors that can be modeled but are often excluded such as external events and accidents at low power and shutdown. It also includes considerations for which methods of analysis have not been developed, for example, operator errors of commission, heroic acts, and influences of organizational performance cannot now be explicitly assessed. Finally, it includes initiators and accident scenarios that have not been conceived.

Incompleteness in a PRA can be addressed for those scope items for which methods are in principal available, and therefore some understanding of the contribution to risk exists. This may be accomplished, by supplementing the analysis to enlarge the scope, using more restrictive acceptance guidelines, or by providing arguments that, for the application of concern, the out-of-scope contributors are not significant. Defense-in-depth is used to compensate for other completeness issues.

4.4 Risk Impacts of Changes

The appropriate numerical measures to use in comparing PRA results to the quantitative guidelines in Figure 3-1 are mean values. The mean values referred to are those that result from the propagation of distributions assigned to uncertain input parameters (and occasionally to alternative models). Methods for propagating input parameter distributions have been developed and, except for dispersion and health effects models, were applied in the NUREG-1150 risk assessments. The resulting uncertainties are large, exceeding two orders of magnitude from the 5-th to 95-th percentile on core damage frequency. The spread in CDF results from the IPEs is generally consistent with the NUREG-1150 uncertainty estimates. As previously mentioned, uncertainties pertaining to phenomenological models tend to increase as accident scenarios progress. In many cases, this leads to significant uncertainties in containment failure probabilities. As part of the NUREG-1150 effort, formal expert elicitation methods were used to quantify key phenomenological uncertainties. Except where significant subsequent research has been conducted, the NUREG-1150 results generally provide the best available quantifications of such uncertainties.

Guidance regarding the treatment of uncertainties will evolve as the Option 3 study progresses; current perspective is provided below by considering a series of questions:

4.4.1 How Risk-Significant Will the Changes Be?

For each affected class of nuclear power plants, the impact of a contemplated regulatory change will be examined relative to the quantitative guidelines in Figure 3-1. The impacts on CDF and LERF are good indicators of impacts on latent-cancer and acute-fatality risks, respectively. Conceptually, averaged over all plants in a class, three possible outcomes can be envisioned for each risk measure. The measure may decrease relative to its quantitative guideline, the impact on the guideline may be indeterminant, or the measure may increase relative to its quantitative guideline.

It is envisioned that most changes would have a major impact on only one of the strategy columns of Figure 3-1. It is unlikely, but conceivable, that a proposed change could result in mixed impacts, for example, decrease CDF while increasing LERF or vice versa. In such cases, for the discussion that follows, the impact on risk is taken to be that on CDF. This is because change in CDF is used to classify risk decreases in the Regulatory Analysis Guidelines and risk increases in Regulatory Guide 1.174. The change in CDF is also a good indicator of societal costs associated with a change.

Risk Decreases

Qualitative arguments may suffice to demonstrate risk would decrease for a particular class of plants as a result of a proposed risk-informed regulatory change.

Changes that would decrease risk, but impose additional licensee burden will be included in risk-informed alternative regulations without detailed value impact analysis because compliance with the alternative regulation is voluntary; that is, licensees may, if they choose, continue to comply with the existing regulation. However, the reasonableness of the additional burden versus the risk decrease will be considered.

There is little point in developing an alternative that no licensee will choose. If the magnitude of the decrease in CDF passes the safety goal screening criteria of the Regulatory Analysis Guidelines, and the change has the potential to pass a value impact analysis, it will be referred to the Generic Safety Issues program for potential mandatory implementation. In particular, changes that would decrease core damage frequencies by greater than 10^{-5} per year while reducing licensee burdens would be referred.

Considerations of uncertainty regarding risk decreases in the Option 3 study must be sufficient to demonstrate that nothing has been overlooked that would actually result in a risk increase.

Risk Impact Indeterminate

Generally if it cannot be determined whether a contemplated change to an existing regulatory requirement would result in a risk increase or a risk decrease, the change would not be risk-informed. But, if it can be demonstrated that the absolute magnitude of the impact would be very small (less than 0.1% of any quantitative guideline) and licensee burden reduction would exceed the dollar value of a 0.1% increase, the option may be included as part of a risk-informed alternative regulation.

Risk Increases

As stated in Section 3, the quantitative guidelines in Figure 3-1 reflect a desired level of safety against which industry averaged risk measures can be compared; a level that is "safe enough" based on the Safety Goal Policy Statement while providing reasonable balance among the defense-in-depth strategies. Changes to existing regulatory requirements should not, therefore, lead to risk increases that go beyond the level of safety implied by the quantitative guidelines.

In principle, if each plant had a high-quality, full-scope, Level 2 PRA with quantitative treatment of uncertainties, the industry-wide impact of alternatives offered under Option 3 could be tracked. In this case, the risk increase (if any) associated with the next alternative could be set relative to the current industry-wide risk profile. This is not a realistic possibility, at least not in the time frame of the initial Option 3 efforts.

Uncertainties must be assessed in making a determination that increases in core damage and large-early-release risk measures would be ~10% or less of the quantitative guidelines. It is anticipated that results from existing PRAs and IPEs coupled with bounding analyses will suffice for this purpose for many cases. Licensee analyses per RG 1.174 may provide a good starting point for assessment of industry-wide risk impacts of some small changes.

As a general principle, changes to existing regulations that would result in risk increases will be avoided if the magnitude of the risk increase is difficult to quantify and little associated NRC or licensee burden reduction would accrue. Where there is potential for burden reduction, that potential should be substantial enough to justify the magnitude of the risk increase.

4.4.2 How Will Initiating Events be Classified (Infrequent versus Rare)?

In assessing whether a particular type of initiating event should be considered rare consideration will be given to the design-basis initiating events postulated in licensee's safety analysis report and other initiating events, both internal and external, identified in PRAs.

Where possible, probabilistic models of initiating event frequencies, using data based on observed occurrence rates to the extent possible, will be utilized.

Models of initiating event frequencies and the parameters of these models will be analyzed to assure that the mean frequency of occurrence of all internal and external initiating events classified as rare does not exceed the 10^{-5} per year guideline. With a high level of confidence, the uncertainty associated with any single parameter or other plausible model choice should not cause this guideline to be exceeded.

It should be noted that the 10^{-5} per year guideline for the collective frequency of rare initiating events includes both internal and external initiating events.

If, based on the preceding considerations, modifying an existing design-basis initiator is contemplated, the potential impact of the change on plant risk measures would, of course, have to be assessed as described in the preceding subsections. For example, it has been argued that very large pipe breaks should be excluded from consideration under the ECCS acceptance criteria of 10 CFR 50.46 because data and fracture-mechanics analyses indicate their frequency of occurrence is very low. Before making such a change to an existing regulatory requirement, the risk impact of plant changes that might result would have to be assessed.

5.0 IMPLEMENTATION OF FRAMEWORK

As stated in the introduction, the framework will be used to guide efforts to develop risk-informed changes to the technical requirements of the regulations in 10 CFR 50. Through implementation of the framework, it is anticipated that Phase 1 of the Option 3 study will identify existing requirements that:

- will be retained
- can be eliminated
- will be revised, enhanced or replaced

In implementing the framework (with its quantitative guidelines), three major steps are followed as depicted in Figure 5-1. The process begins with the selection and prioritization of the regulations in 10 CFR 50 to be risk-informed as discussed in Section 5.1. After a regulation is selected and its technical bases are studied, a risk-informed alternative to the technical requirements of that regulation will be developed as discussed in Section 5.2. In the third step, which is discussed in Section 5.3, an evaluation is performed of the risk-informed alternative.

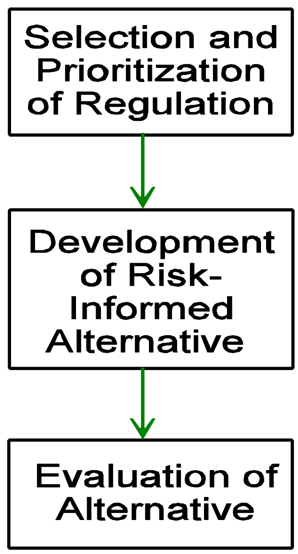

Figure 5-1 Approach for selection, development, and evaluation of risk-informed alternative.

5.1 Step 1: Select and Prioritize Regulations to be Risk Informed

The first major element in the process is the selection of the regulation that needs to be risk-informed. The selection and prioritization process consists of five major components as shown in Figure 5-2: a coarse screening of the regulations in 10 CFR Part 50, a safety concern screening to identify "holes" in the regulations, a second screening to determine if a regulation even warrants risk-informed change, a linking to identify ties to other regulations or implementing documents, and a prioritizing of the regulations to be risk-informed.

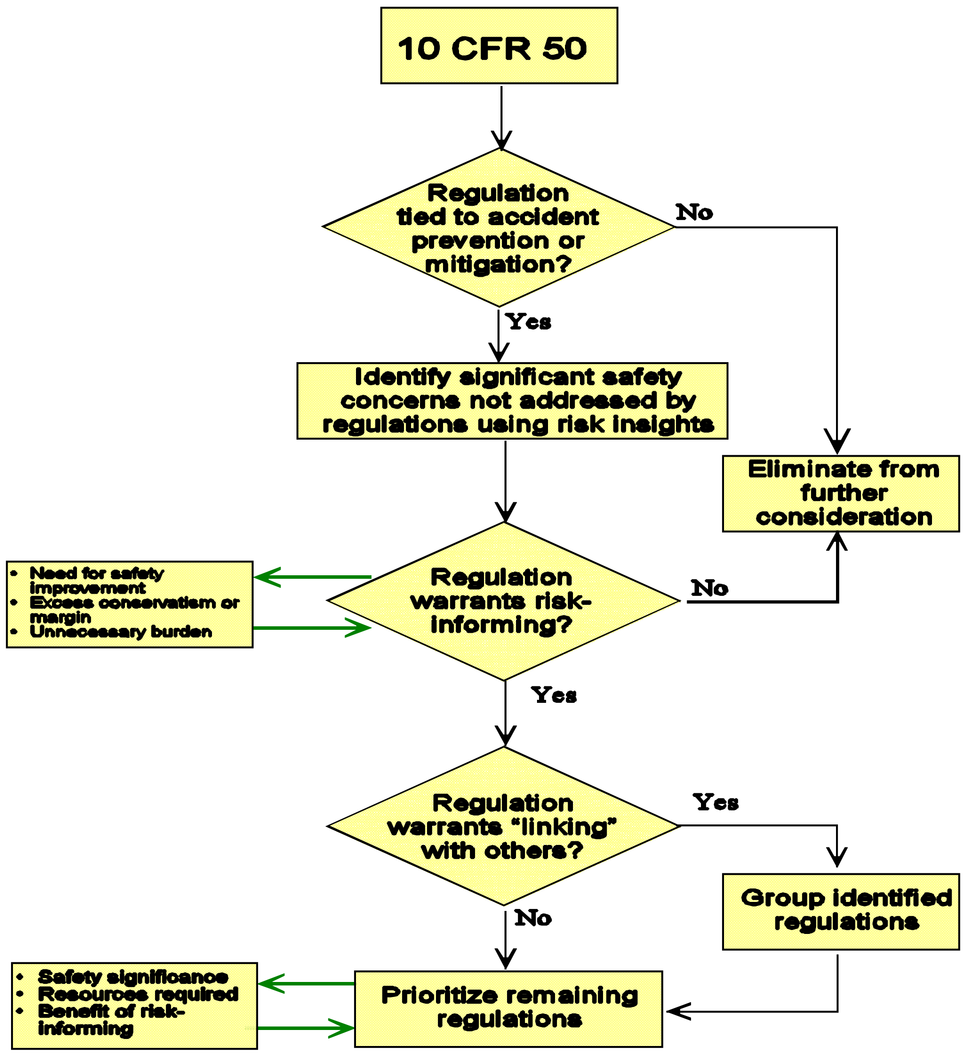

Figure 5-2 Process for selecting and prioritizing regulations to be risk-informed.

Coarse Screening of 10 CFR 50

As indicated previously, this framework is directed toward existing regulations that have an impact on prevention or mitigation of core-damage accidents, because these accidents present the most risk to the public and risk information is most prevalent for such accidents. In the future, the framework can be adapted and extended to apply to regulatory requirements that impact non-core-damage accidents.

A preliminary coarse screening was conducted of Parts 50 and 100, and each regulation was placed in one of two bins:

1. Regulations that do not have an impact on prevention or mitigation of core-damage accidents. These consist of sections that are purely procedural or provide legal or technical definitions, refer to enforcement provisions and/or penalties for misconduct, concern financial and insurance requirements, specify routine exposure limits from plant operation, pertain to decommissioning, or impact only non-core-damage accidents.

2. Regulations that could potentially impact prevention or mitigation of core-damage accidents.

The results of the preliminary screening are presented in Appendix A. Many of the regulations falling into Bin 1 are process-oriented. Although not themselves candidates for risk-informed changes, it is conceivable that some process-oriented regulations may have to be changed for the sake of consistency due to risk-informed changes made to regulations in Bin 2. Bin 2 includes all of the possible candidates to be risk informed identified in a recent Nuclear Energy Institute (NEI) letter. The prime candidates identified by NEI for risk-informed assessment and change are (Ref. 14):

- LOCA, ECCS analyses, 10 CFR 50.46 and Appendix K to Part 50

- Codes and Standards, 10 CFR 50.55a

- GDC 4, Appendix A to Part 50 and associated regulatory guidance documents that are linked to pipe-whip and dynamic effects

- Environmental qualification of electric equipment important to safety for nuclear power plants, 10 CFR 50.49

- Standards for combustible gas control system in light-water-cooled power reactors, 10 CFR 50.44

- GDC 19, Appendix A to Part 50 and associated regulatory guidance documents linked to control room ventilation

- GDC 17, Appendix A to Part 50 and associated guidance documents related to electrical power systems

Safety Concerns Not Addressed in 10 CFR 50

In the process of making risk-informed changes to the existing regulations, it is also important to identify risk-significant safety issues not explicitly addressed in current regulations. At a very coarse level, an attempt has been made to find issues that are important to accident risks, in terms of accident types, which are not addressed in the current Part 50 regulations. Table 5-1 shows a mapping of accident types that are important to CDF or LERF to Part 50 regulations. Further investigation is necessary in order to identify whether there are major risk contributors associated with these accident types that need to be addressed by the regulations.

Table 5-1 Regulatory Coverage of Some Accidents Important to Risk (Preliminary)

Accident Types Important to CDF/LERF	Regulations in Part 50
SBO	50.63, 50.34 (f) (ix)
ATWS	50.62
LOCAs	50.34 (f) (iv) - Small Break LOCA, 50.46 - ECCS Acceptance Criteria, App. K, App. J
Transients with DHR Loss	50.34 (f) (i) - DHR Reliability
Transients with Injection Loss	50.34 (f) (v), 50.34 (f) (vii), 50.34 (f) (viii), 50.34 (f) (x), 50.34 (f) (xi)

Table 5-1 Regulatory Coverage of Some Accidents Important to Risk (Preliminary)

Accident Types Important to CDF/LERF	Regulations in Part 50
Early Containment Failure	50.34 (f) (xii), 50.44 - H2 control, App. A
Containment Bypass-ISLOCA/SGTR	App. A (very indirectly)
Loss of Containment Isolation	App. A
Internal Fire	App. R
Internal Flood	
External Events	(Part 100 for siting), App. S
Events at Low Power and Shutdown	

One feature that is immediately obvious from the table is the fact that many of the risk-significant accident types are only covered by 50.34 (f) (..), the "TMI-related regulations." This set of regulations applies only to plants whose license applications were pending as of February 1982. (The paragraph under 50.34 (f) identifies a specific set of plants to which these rules were applicable; none of these plants have been constructed.) By inference, these regulations do not apply to the current set of operating plants, so there is, in principle, the possibility that one or more risk-significant safety issues may need to be assessed in the risk-informed process.

Some risk-significant accident types and related events do not find any mention in the current regulations. Except for hydrogen, threats posed by severe accidents are not specifically mentioned in existing regulations. Often, one has to "stretch" the rather general language contained in the regulation to infer its applicability to a particular accident class. An example would be interpreting the contents of Appendix A to cover the containment bypass accident category.

Second Screening

As indicated in Figure 5-2, a second screening is performed to identify those regulations that do not warrant risk-informed changes and can be eliminated from further consideration because (1) there is no need for safety improvement, (2) there is no excess conservatism or margin in the regulation's technical requirements, and (3) there is no unnecessary burden associated with the the technical requirements of the regulation.

Any regulation for which a safety enhancement may be necessary, based on the quantitative guidelines presented in Figure 3-1, will clearly need to be retained and prioritized for risk-informed changes. For those regulations for which no safety enhancement is deemed necessary for its technical requirements, given that licensees will have the option of choosing between an existing regulation and its risk-informed counterpart, there is little purpose in promulgating a risk-informed regulation that does not offer a significant tangible benefit to at least some licensees. Accordingly, only those regulations whose technical requirements (of this latter category) which result in unnecessary burden reduction will be retained and prioritized for risk-informed changes.

Linking

Further evaluation of the remaining regulations is performed to identify any ties, overlaps or redundancies to determine if sets of existing regulations should be "linked or grouped" for further risk-informed study. There are instances in the current 10 CFR 50 where a particular aspect of plant design, construction or operation is addressed in more than one regulation or associated implementing document. In these instances, it may be more efficient and effective to address all of the impacted regulations together as a single group. In particular, linking regulations will help to avoid (or at least be cognizant of) situations where a particular technical requirement may be modified or eliminated in a risk-informed regulation, but that same technical requirement is still specified, as it currently exists, in another regulation (or associated implementing document). In the discussion that follows the singular use of the term regulation should be understood to apply to such linked sets of regulations.

Prioritization

The regulations that survive the secondary screening are prioritized. The highest priority candidates are selected for detailed evaluation in Step 2. Three factors are considered in prioritizing candidate regulations to be risk informed:

- the safety significance of each regulation,

- the potential resources required to risk inform (considering complexity, information requirements, need for a demonstration plant, time, manpower, etc.), and

- the benefit of making risk-informed changes to the regulation (e.g., the potential for reducing unnecessary burden).

In assessing safety significance, both the impact of a regulation on the quantitative guidelines in Figure 3-1 and the number of plants affected by the regulation will be considered. It is generally straightforward to determine which, if any, of the four high-level defense-in-depth strategies a regulation impacts. The safety significance of the impact can, in some cases, be characterized qualitatively. In other cases simple quantitative analyses of the contributions from accident scenarios impacted by the regulation may be performed based on available IPEs and PRAs.

5.2 Step 2: Development of Risk-Informed Changes

The second major element in the process is to develop the risk-informed changes to the technical requirements for the high-priority regulations identified in Step 1. Two approaches are followed for developing risk-informed changes to a regulation. Both approaches begin with an examination of the concern or concerns that necessitated the regulation, and both approaches have the same overall objective, which is to develop risk-informed requirements for dealing with the identified concern.

One approach starts from the current set of technical requirements of the regulation and attempts to develop risk-informed changes by analyzing the technical requirements. The second approach takes a fresh start by applying the four high-level defense-in-depth strategies; in effect, ignoring the existing technical requirements of regulation.

There are two principal reasons for following two approaches to developing a risk-informed alternative to a regulation. The first reason is for completeness. Following both of the above approaches gives greater confidence that all reasonable risk-informed options have been identified. The second reason is to identify a risk-informed alternative that is the most optimal by looking at the concern from an alternative perspective, that is, without being constrained, or unduly influenced, by the existing requirements.

Potential changes identified by either of these two approaches are developed based on the following six considerations:

- risk insights from plant specific PRAs

- industry experience

- consistency with the quantitative

guidelines identified in the framework document

- reasonable cost burden
- proven technology
- suitability for performance-based compliance monitoring

The potential changes derived from both approaches are evaluated to arrive at the risk-informed alternative.

5.2.1 Revising Current Requirements Approach

The approach based on revising the existing technical requirements is shown in Figure 5-3. Each of the six steps in this approach is described below.

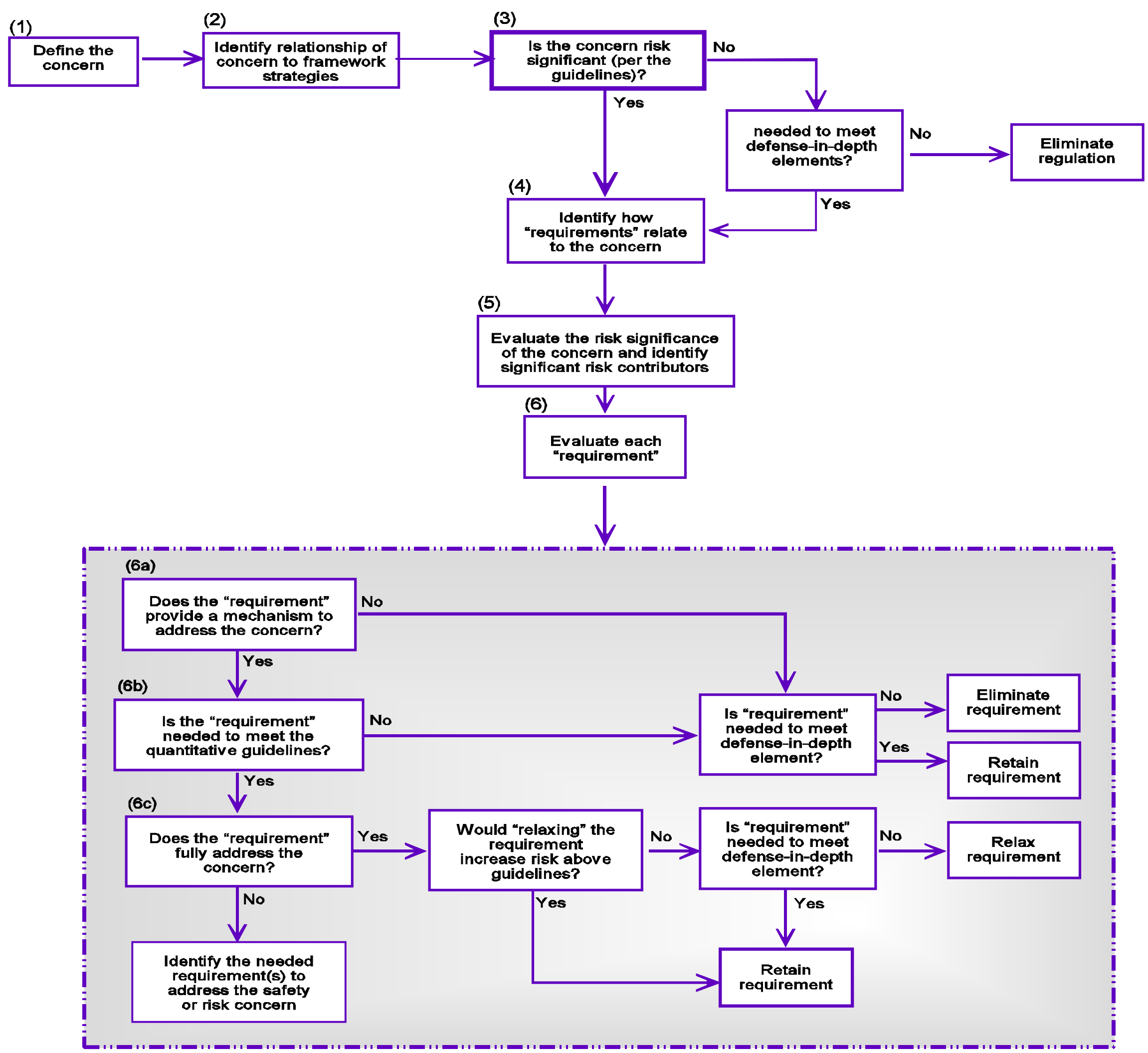

Figure 5-3 Current Requirements Approach to Develop Risk-Informed Changes

(1) Define the concern:

As mentioned previously, development of risk-informed changes to the technical requirements of a regulation begins with an examination of the concern or concerns that necessitated the regulation. Only after the concern is clearly understood, can a determination be made as to how risk-significant the concern is, and how effectively the concern is addressed by the existing requirements. The concern should be expressed in terms of its risk significance (e.g., which risk-significant accidents are impacted, and how significant is this impact).

(2) Identify relationship of concern to framework strategies:

In Section 2 of this report, four defense-in-depth strategies to be considered in making risk-informed change to the regulations were identified. Two of these strategies are preventive (limit frequency of accident initiators and limit probability of core damage given an initiator), and two of the strategies are mitigative (limit radionuclide releases given core damage and limit public health effects given release). The next step in developing risk-informed changes is to identify which of the four strategies are impacted by the concern.

(3) Is the concern risk significant (per the guidelines)?

The risk significance of the concern is assessed against the quantitative guidelines in Figure 3-1. Based on information derived from PRAs, an assessment of the quantitative significance of the concern is made with respect to the quantitative guidelines presented in Figure 3-1 for various types of plants (as defined by their nuclear steam supply systems or containment designs). If the risk significance of the concern results in values significantly below the quantitative guidelines, then the regulation (in its entirety) may become a candidate for elimination. Such regulations must be evaluated to determine (1) if the low risk is because of the technical requirements imposed by the regulation, and if not, then (2) whether the technical requirements are needed to meet any of the defense-in-depth elements. If it is determined that they are not needed to meet the guidelines nor are they needed to maintain defense-in-depth, then the regulation itself becomes a candidate for elimination. It is important to note that all candidates for elimination identified through this process (which is derived with basis on the four reactor safety cornerstones) will also be examined to assure that their elimination will not have any adverse impact on AOOs and the radiation safety and security cornerstones.

(4) Identify how "requirements" relate to the concern:

Each technical requirement contained in the existing regulation is identified and described in detail in terms of the affected systems, structures, components and procedures (if any) for the various types of plants and the criteria used for assessing compliance with the requirements. A review is then made to determine the relationship of each requirement to other regulations and implementing documents, such as regulatory guides, standard review plan, branch technical positions, generic letters, etc. The purpose of this review is to obtain a detailed understanding of the implications of revising any particular requirement in terms of its impact across the body of the regulations and implementing documents.

Subsequent to the above review, the basis and method of implementation of the requirements by industry are identified and described. A determination is made as to whether the requirement has been implemented by the licensees on the basis of the regulation alone, on the basis of an associated regulatory guide or other implementing document, or on some other basis.

Lastly, each requirement identified at the beginning of this step is evaluated in the context of how effectively it addresses the defined concern.

(5) Evaluate the risk significance of the concern and identify significant risk contributors:

This step is essentially a detailed extension of step (3), above. In step (3), the risk significance of the concern was evaluated, at a high level, in comparison with the quantitative guidelines provided in Figure 3-1. Given that the concern was determined to be risk-significant in step (3), in this step, available PRA information (e.g., NUREG-1150, or IPEs) is reviewed to determine what is driving the risk-significance of the concern. For the various types of plants (as defined by their nuclear steam supply systems or containment designs), the risk significant contributors are identified, where possible, in terms of the PRA results (e.g., dominant accident sequences, or dominant containment failure modes).

(6) Evaluate the each requirement:

In this step, each technical requirement identified in step (4) is evaluated to determine if, and how, it should be risk-informed. Options for risk-informed changes to the technical requirements broadly fall into the following three categories:

- eliminate the current requirement
- retain the current requirement
- revise, enhance, or supplant the current requirement

Guidance as to which category each requirement falls into is provided by answering the three questions described below.

6a Does the requirement provide a mechanism to address the concern?

The answer to this question should have been obtained during step (4) above. If the requirement does not provide a mechanism to address the concern, then it should be evaluated to determine whether it is needed to meet any of the defense-in-depth elements. If it is determined that the requirement is needed to maintain defense-in-depth, then the requirement is retained. However, if it is determined that the requirement is not needed to maintain defense-in-depth, then it becomes a candidate for elimination. It is important to note, as before, that all requirements that are identified as candidates for elimination through this process will also be examined to assure that their elimination will not have any adverse impact on the radiation safety and security cornerstones.

If the requirement does provide a mechanism to address the concern, then it is subjected to the following question.

6b Is the requirement needed to meet the quantitative guidelines?

Based on information obtained in steps (3-5), a determination is made as to whether the requirement is necessary in order for the strategies impacted by the concern to meet the associated quantitative guidelines provided in Figure 3-1. If the requirement is determined not to be necessary to meet the quantitative guidelines, then it will be either eliminated or retained based on whether it is needed to meet any of the defense-in-depth elements, as discussed for the previous question. If the requirement is determined to be necessary to meet the quantitative guidelines, then it is subjected to the following question.

6c Does the requirement fully address the *concern?*

This question is used to determine whether or not a safety enhancement would be appropriate. It is possible that there are aspects of the defined concern which are not fully addressed by the existing requirement (or requirements). In this case, any necessary additional requirements should be identified, so that the concern will be fully addressed.

If the requirement does fully address the concern, then the requirement is evaluated to determine whether or not it can be relaxed and still maintain risk below the quantitative guidelines. If relaxing the requirement would increase risk above the guidelines, then the requirement is retained, as is. If relaxing the requirement would still maintain risk below the guidelines, then it can be relaxed, as long as it is not needed to meet any of the defense-in-depth elements, as discussed previously.

5.2.2 Developing Alternative Requirements Approach

As noted above the main difference between the two approaches to developing risk-informed changes is that risk-informed changes obtained through implementation of the alternative requirements approach are developed without reference to the existing technical requirements of the regulation. In this approach, as seen from Figure 5-4, risk-informed changes for addressing the concern can be identified during any of steps (2-4). This allows changes to be developed from different perspectives. The four steps in an alternative approach that begins afresh from a risk-informed perspective are described below using the four strategies of the framework and with again defining the concern.

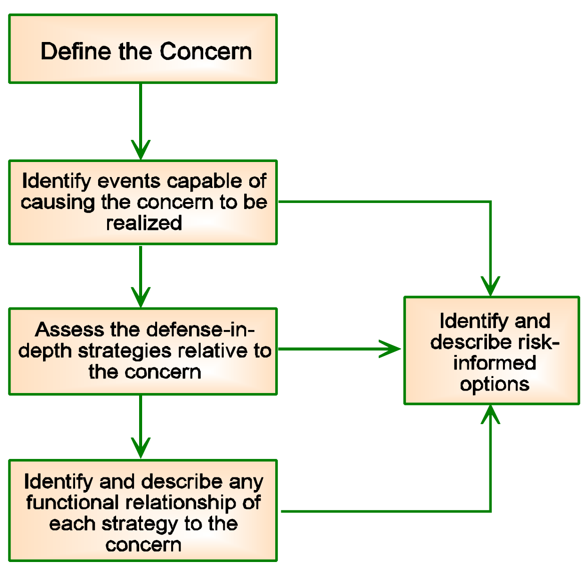

Figure 5-4 Alternative requirements approach to developing risk-informed options

(1) Define the concern:

This step is very similar to step (1) for the revising current requirements approach. As mentioned previously, development of risk-informed changes to a regulation begins with an examination of the concern or concerns that necessitated the regulation. The concern should be expressed in terms of its risk significance (e.g., which risk-significant accidents are impacted, and how significant is this impact).

(2) Identify events capable of causing the concern to be realized:

After the concern is defined, an identification is made at a high-level of events that could cause the concern to be realized. For example, if the concern is that a deflagration/detonation of combustible gas could threaten containment, for the concern to be realized there must be generation of combustible gas from metal-water reactions during an accident in which significant core damage occurs. If the concern is that rupture of a large pipe in the reactor coolant system could threaten public health and safety, for the concern to be realized emergency core cooling and containment functions would also have to fail. Existing PRAs can, generally, provide more specific insights regarding specific sequences of events that are most likely to cause an identified concern to be realized.

(3) Assess the defense-in-depth strategies relative to the concern:

As mentioned previously, Section 2 of this report identifies four defense-in-depth strategies for limiting accident risk. Three of these strategies also have quantitative guidelines associated with them, as shown in Figure 3-1. In this step, the efficacy of each strategy relative to preventing and mitigating the identified concern is assessed. For those strategies that address the concern, performance-based options can be developed with high-level acceptance criteria, which would allow licensees substantial flexibility in meeting them. In addition, if it is anticipated that it may be difficult for licensees to meet the high-level acceptance criteria based on the strategies that address the concern, similar type options can be developed based on the remaining strategies. For example, the reduction of the frequency of an accident class under which the concern becomes less manageable may be more practical than ensuring the operability of a mitigating system under the same conditions.

(4) Identify and describe any functional relationship of each strategy to the concern:

Understanding the functional relationships between each strategy and the concern allows practical methods of applying each defense-in-depth strategy to the defined concern to be identified, for relevant plant types. These changes are expected to be much more prescriptive than those developed under the preceding step. For example, specific hardware or procedures may be identified in these changes for applying a specific strategy to the concern. As in the previous step, the changes may relate to the strategies that address the concern, or it may prove to be more practical to develop changes related to the other strategies. For example, station blackout accidents may impose the most severe conditions on the plant's ability to successfully control combustible gas concentrations. An option reducing the frequency of station blackout may prove to be more practical for managing the defined concern than attempting to ensure that mitigating systems can successfully operate under station blackout (SBO) conditions.

5.3 Step 3: Evaluation of Risk-informed Alternative

In the previous step, all changes were developed based on safety and risk implications with consideration of the defense-in-depth elements. These changes were evaluated to arrive at a risk-informed alternative to an existing regulation. In this step, the risk-informed alternative is evaluated in order to estimate the associated NRC and licensee burdens, for both implementing and applying the alternative, and to compare these estimates with similar estimates for the existing regulation. The factors affecting both NRC and licensee burden are provided below.

Factors impacting NRC:

- **Need for a rule change —** The formal rule-making process can involve a substantial expenditure of resources by the NRC. Therefore, whether or not a proposed risk-informed alternative necessitates a change to the regulation itself is an important consideration in determining the NRC burden.

- **Impact on other regulations —** Due to the interrelationship of various regulations, changes to one regulation may require corresponding changes in other regulations, which can increase the burden to the NRC. Regulations that do not have a relationship with other regulations can be addressed unilaterally in the risk-informed process.

- **Need to revise or modify implementing documents —** In order to implement an option, it may be necessary to revise or modify one or more implementing documents (e.g., regulatory guides or standard review plan sections). Modifications to the implementing documents may represent the sole change associated with the risk-informed alternative, or these changes may be in conjunction with changes to the regulation (or regulations).

- **Need to create a new implementing document —** In some instances, a new implementing document may need to be developed. Development of the implementing document may or may not be in conjunction with changes to the regulation (or regulations), and may or may not be in conjunction with modifications to other implementing documents.

- **Extent of regulatory analysis required —** The extent of regulatory analysis required in support of a risk-informed alternative may range from virtually none, if existing information and analysis results satisfactorily address the safety benefit and NRC and licensee burdens associated with the risk-informed alternative, to substantial, if significant resources need to be expended to evaluate previously unanalyzed aspects of the risk-informed alternative .

- **Need for NRC review of licensee submittals —** If the particular aspects of a risk-informed alternative require that each licensee provide a submittal to the NRC, then the associated NRC review costs need to be considered.

- **Impact on NRC inspection activities —** Consideration needs to be given to the impact, if any, that a particular risk-informed alternative has on NRC inspection activities. The nature of this impact may be to increase the burden associated with NRC inspection activities, or to decrease the burden.

Factors impacting Licensees:

- **Need for new or modified equipment —** As a result of a particular option, the need for the licensee to remove, install, replace or modify existing plant equipment can be a contributor to licensee burden. In some cases, replacement of equipment (when necessary) may result in a decrease in licensee burden, if the risk-informed alternative allows replacement equipment of a lower pedigree than the existing equipment.
- **Need for analysis —** Consideration is given to the need for, and extent of, any

analysis required to be performed by the licensee. For example, if use of a PRA is required, then there may be burden associated with modifying the PRA to meet a given level of completeness and confidence. Also, consideration needs to be given to the burden associated with documentation and reporting requirements associated with the specified analysis.

- **Impact on maintenance and inspection activities —** Consideration needs to be given to the impact, if any, that a particular risk-informed alternative has on licensee inspection and maintenance activities. The nature of this impact may be to increase or decrease the burden associated with these activities.

- **Impact on technical specifications —** Consideration needs to be given to the impact, if any, that a particular risk-informed alternative has on plant technical specifications. This impact, which can either increase or decrease burden, may involve such things as system or equipment testing frequencies, or conditions for which the plant must shut down.

- **Impact on procedures and training —** Consideration needs to be given to the impact, if any, that a particular risk-informed alternative has on plant procedures and training. If a particular risk-informed alternative requires plant procedures to be changed or written, consideration must be given to the cost of modifying or writing the procedures, as well as to the cost of the associated operator training to become familiar with the new procedures.

6.0 SUMMARY

This document presents a framework and guidelines to be used in making risk-informed changes to the existing technical requirements of 10 CFR 50. The approach maintains four high-level defense-in-depth functions, which support the protection of the public health and safety goal and are consistent with the reactor safety cornerstones developed for regulatory oversight. Risk information is used to evaluate the effectiveness of the defense-in-depth approach. Although regulations will be revised or originated based on risk information, they will retain deterministic characteristics. The development of risk-informed regulatory requirements will be guided by quantitative safety objectives, insights derived from PRAs and IPEs, and the need to account for uncertainty, particularly in cases where one or more of the high-level defense-in-depth functions is precluded.

7.0 REFERENCES

1. USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.

2. USNRC, SECY-98-300, "Options for Risk-informed Revisions to 10 CFR 50 - 'Domestic Licensing of Production and Utilization Facilities,'" December 23, 1998.

3. USNRC, SECY-99-264, "Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50," November 8, 1999.

4. USNRC, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, NUREG/BR-0058, November, 1995.

5. USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.

6. USNRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

7. USNRC, "New NRC Reactor Inspection and Oversight Program," NUREG-1649, Rev. 1, May 1999.

8. Letter from D. A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to The Honorable Shirley Ann Jackson, Chairman, U. S. Nuclear Regulatory Commission, "The Role of Defense in Depth in a Risk-Informed Regulatory System," May 19, 1999.

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10. USNRC, "New NRC Reactor Inspection and Oversight Program," NUREG-1649, Rev. 1, May 1999.

11. USNRC, SECY-99-007, "Recommendations for Reactor Oversight Process Improvements," January 8, 1999.

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13. E. B. Haugen, *Probabilistic Approaches to Design*, John Wiley & Sons, Inc., London, 1968.

14. Letter from Joe F. Colvin, President, Nuclear Energy Institute, to Richard A. Meserve, Chairman, U.S. Nuclear Regulatory Commission, January 19, 2000.

APPENDIX A

COARSE SCREENING OF 10 CFR 50 AND 100

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APPENDIX A

A.1 Coarse Screening of Part 50

Based on a preliminary review, out of a total of 82 regulations and 17 published Appendices under Part 50, the following 57 regulations and 8 Appendices, as listed below in Table A-1, have no relevance to risk-informing.

Table A-1 Part 50 Regulations and Appendices that have no relevance to Risk-Informing

General Provisions	
50.1	Basis, purpose, and procedures applicable
Requirement of License, Exceptions	
50.10	License required.
50.11	Exceptions and exemptions from licensing requirements.
50.13	Attacks and destructive acts by enemies of the United States; and defense activities.
Classification and Description of Licenses	
50.20	Two classes of licenses.
50.21	Class 104 licenses; for medical therapy and research and development facilities.
50.22	Class 103 licenses; for commercial and industrial facilities.
50.23	Construction permits.
Applications for Licenses, Form, Contents, Ineligibility of Certain Applicants	
50.30	Filing of applications for licenses; oath or affirmation.
50.31	Combining applications.
50.32	Elimination of repetition.
50.33a	Information requested by the Attorney General for antitrust review.
50.34a	Design objectives for equipment to control releases of radioactive material in effluents -- nuclear power reactors.
50.35	Issuance of construction permits.
50.36a	Technical specifications on effluents from nuclear power reactors.
50.36b	Environmental conditions.
50.37	Agreement limiting access to Restricted Data.
50.38	Ineligibility of certain applicants.
50.39	Public inspection of applications.

Table A-1 Part 50 Regulations and Appendices that have no relevance to Risk-Informing

Standards for Licenses and Construction Permits	
50.40	Common standards.
50.41	Additional standards for class 104 licenses.
50.42	Additional standards for class 103 licenses.
50.43	Additional standards and provisions affecting class 103 licenses for commercial power.
50.45	Standards for construction permits.
Issuance, Limitations, and Conditions of Licenses and Construction Permits	
50.50	Issuance of licenses and construction permits.
50.51	Continuation of license.
50.52	Combining licenses.
50.53	Jurisdictional limitations.
50.55	Conditions of construction permits.
50.56	Conversion of construction permit to license; or amendment of license.
50.57	Issuance of operating license.
50.58	Hearings and report of the Advisory Committee on Reactor Safeguards.
50.64	Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors.
Inspections, Records, Reports, Notifications	
50.70	Inspections.
50.71	Maintenance of records, making of reports.
50.72	Immediate notification requirements for operating nuclear power reactors.
50.74	Notification of change in operator or senior operator status.
50.75	Reporting and record keeping for decommissioning planning.
US/IAEA Safeguards Agreement	
50.78	Installation information and verification.
Transfers of Licenses -- Creditors' Rights -- Surrender of Licenses	
50.80	Transfer of licenses.
50.81	Creditor regulations.
50.82	Termination of license.
Amendment of License or Construction Permit at Request of Holder	
50.90	Application for amendment of license or construction permit.
50.91	Notice for public comment; State consultation.

Table A-1 Part 50 Regulations and Appendices that have no relevance to Risk-Informing

Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission	
50.100	Revocation, suspension, modification of licenses and construction permits for cause.
50.101	Retaking possession of special nuclear material.
50.102	Commission order for operation after revocation.
50.103	Suspension and operation in war or national emergency.
Enforcement	
50.110	Violations.
50.111	Criminal penalties.
50.120	Training and qualification of nuclear power plant personnel.
Appendix C:	A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Facility Construction Permits
Appendix F:	Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities (Not relevant to reactors)
Appendix H:	Reactor Vessel Material Surveillance Program Requirements
Appendix I:	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents (applicable to routine emissions)
Appendix L:	Information Requested by the Attorney General for Antitrust Review of Facility License Applications
Appendix M:	Standardization of Design; Manufacture of Nuclear Power Reactors; Construction and Operation of Nuclear Power Reactors Manufactured Pursuant to Commission License
Appendix N:	Standardization of Nuclear Power Plant Designs: Licenses to Construct and Operate Nuclear Power Reactors of Duplicate Design at Multiple Sites
Appendix O:	Standardization of Design: Staff Review of Standard Designs

The remaining 23 regulations and 9 Appendices to Part 50, as listed below in Table A-2, have a potential relevance to the risk-informed process. The relevance, however, may only be indirect or partial in some cases. As mentioned above, a second screening will need to be carried out to determine those that have a direct relevance to accident prevention and/or mitigation.

Table A-2 Part 50 Regulations potentially relevant to Risk-Informing

50.2	Definitions
50.12	Specific exemptions.
50.33	Contents of applications; general information.
50.34	Contents of applications; technical information
50.36	Technical specifications.
50.44	Standards for combustible gas control system in light-water-cooled power reactors.
50.46	Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.
50.47	Emergency plans.*
50.48	Fire protection.*
50.49	Environmental qualification of electric equipment important to safety for nuclear power plants.
50.54	Conditions of licenses.
50.55a	Codes and standards.
50.59	Changes, tests and experiments.
50.60	Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.
50.61	Fracture toughness requirements for protection against pressurized thermal shock events.*
50.62	Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.
50.63	Loss of all alternating current power.
50.65	Requirements for monitoring the effectiveness of maintenance at nuclear power plants. (Eff. July 10, 1996)
50.66	Requirements for thermal annealing of the reactor pressure vessel.
50.68	Criticality accident requirements.
50.73	License event report system.
50.92	Issuance of amendment.
Appendix A:	General Design Criteria for Nuclear Power Plants
Appendix B:	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
Appendix E:	Emergency Planning and Preparedness for Production and Utilization Facilities (Partly relevant if EP for advanced reactors is different based on risk)
Appendix G:	Fracture Toughness Requirements (Maybe relevant)
Appendix J:	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
Appendix K:	ECCS Evaluation Models
Appendix Q:	Pre-application Early Review of Site Suitability Issues (Partly relevant)
Appendix R:	Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979
Appendix S:	Earthquake Engineering Criteria for Nuclear Power Plants (Partly relevant)

*50.47, 50.48 and 50.61 are not part of the scope of this effort; these regulations are being addressed under other programs.

A.2 Coarse Screening of Part 100

Part 100, “Reactor Site Criteria”, deals with the factors that influence the approval of a site for locating and constructing a nuclear power plant and the criteria to be used in arriving at a decision on site selection. Public health and safety and the development of emergency plans are cited as important criteria in this regard and it is stated in Part 100.1 “Purpose” that “the primary factors that determine public health and safety are reactor design, construction, and operation.” This implies that the provisions of Part 100, like those of Part 50, are also suitable candidates for risk-informing.

Part 100 was significantly revised in December 1996 to reflect information derived from risk-oriented studies of the siting of nuclear power plants [4,5]. Other objectives [6] of the revision were to provide a stable regulatory basis for seismic and geologic siting and applicable earthquake engineering design of future nuclear power plants and to decouple siting criteria from design by relocating the source term and dose requirements that apply primarily to plant design into Part 50.

Part 100.1 “Purpose”, Part 100.2 “Scope”, and Part 100.3 “Definitions” were changed. A new Subpart B “Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997" was added consisting of new paragraphs 100.20 “Factors to be considered when evaluating sites”, 100.21 “Non-seismic siting criteria” and 100.23 “Geologic and seismic siting criteria”. (However, the old Appendix A to Part 100 that defined the seismic and geologic siting criteria earlier has been retained unchanged. It is stated to apply to an operating license applicant or holder whose construction permit was issued prior to January 10, 1997). The older Parts 100.10 and 100.11 are now placed under Subpart A “Evaluation Factors for Stationary Power Reactor Site Applications Before January 10, 1997 and for Testing Reactors”.
The same criteria that were applied to the Part 50 regulations have been applied to Part 100 to subdivide them into the 2 categories, Category 1 and Category 2, defined above. It should be noted that Parts 100.10, 100.11 of Subpart A and Appendix A to Part 100 now refer specifically to power reactor site applications received before January 10, 1997 or to testing reactors. If there are no operating license applicants or holders with construction permits issued prior to 1/10/97 who are currently seeking approval for a site to construct a commercial nuclear power plant, then it would appear that Subpart A is essentially moot. In principle, Parts 100.10 and 100.11 and Appendix A should be candidates for risk-informing.

Table A-3 Part 100 Regulations that have no relevance to Risk-Informing

100.3	Communications
100.8	Information collection requirements: OMB approval

Table A-4 Part 100 Regulations potentially relevant to Risk-Informing

100.1	Purpose
100.2	Scope
100.3	Definitions
100.10	Factors to be considered when evaluating sites
100.11	Determination of exclusion area, low population zone, and population center distance
100.20	Factors to be considered when evaluating sites
100.21	Non-seismic siting criteria
100.23	Geologic and seismic siting criteria
Appendix A:	Seismic and Geologic Siting Criteria for Nuclear Power Plants

Attachment 2

FEASIBILITY STUDY FOR A RISK-INFORMED ALTERNATIVE TO 10 CFR 50.44 "Standards for Combustible Gas Control System in Light-water-cooled Power Reactors"

Prepared by:

Office of Nuclear Regulatory Research
Division of Risk Analysis and Applications
Probabilistic Risk Analysis Branch

Mary Drouin
Alan Kuritzky
John Lane
Leslie Lancaster

Office of Nuclear Reactor Regulation
Division of Systems Safety and Analysis
Probabilistic Safety Assessment Branch

Michael Snodderly

Brookhaven National Laboratory
John Lehner
Vinod Mubayi
Trevor Pratt

Sandia National Laboratories
Allen Camp
Jeff LaChance

ERI Consulting
Eric Haskin

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1. INTRODUCTION

1.1 Background

In SECY-98-300 [1], the NRC staff presented the following three options for modifying the regulations in 10 CFR Part 50 to make them risk-informed:

1. Continue ongoing rulemaking activities and risk-informed approaches making no changes to the current Part 50
2. Change the special treatment rules in Part 50 to modify their scope to be risk-informed, and
3. Make changes to specific requirements in the body of the regulations, including the general design criteria (GDC).

In a June 8, 1999, Staff Requirements Memorandum (SRM), the Commission:

- Approved proceeding with the current rulemakings in Option 1
- Approved implementing Option 2, and
- Approved proceeding with a study of Option 3

SECY-99-264 [2] provides the NRC staff's plan for the study phase of its work to risk inform the technical requirements of 10 CFR 50 (i.e., Option 3 of SECY-98-300). The plan consists of two phases:

- an initial feasibility study (Phase 1) where recommendations to the Commission on proposed changes will be made, and
- an implementation phase (Phase 2) where changes resulting from Phase 1 approved by the Commission will be made.

Phase 1 consists of three tasks:

Task 1: Identification of candidate changes to requirements and design basis accidents.

- This task provides a first screening of the technical requirements in 10 CFR Part 50, implementing documents, and design basis accidents (DBAs). This screening uses criteria [2] to identify the best candidates for change. The criteria are used to identify requirements and DBAs which appear to have a frequency, risk, or conservatism which is either inordinately high or low.

Task 2: Prioritization of candidate changes to requirements and design basis accidents

- Prioritization criteria used in this task include rough estimates of the values and impacts of the candidate change (including values in safety benefit and burden reduction, and impacts in costs to the NRC and the licensee to make the change); and the practicality of the candidate change.

Task 3: Identification of recommended changes to requirements

- This task will establish the scope and feasibility of implementing the candidate changes identified in the previous tasks.

The changes identified and evaluated in the above tasks can include adding provisions to 10 CFR Part 50 allowing for risk-informed alternatives to the present requirements, revising specific requirements to reflect risk-informed considerations, or deleting unnecessary or ineffective regulations.

However, in addition to the above tasks, SECY-99-264 states that the process for risk-informing will be tested on an expedited basis using at least two examples. In an attachment to SECY-99-264, 10 CFR 50.44 is identified as a candidate for prompt revision based on work done to date and the feedback received from stakeholders.

SECY-00-0086 [3] provides the first status report on risk-informing the technical requirements of 10 CFR PART 50. In SECY-00-0086 the staff describes a number of activities that have been accomplished since the beginning of the study:

- Developed an initial framework for risk-informing 10 CFR Part 50
- Met with stakeholders (both internal and external) to obtain their input on this study.
- Performed a trial implementation (i.e., 10 CFR 50.44) to test the procedures described in SECY-99-264.

The staff developed an initial framework (Attachment 1 to Reference [3]) to more clearly define and guide the work to be performed under the three tasks of Phase 1. The framework employs an approach that builds upon the defense in-depth philosophy and the concept of safety margins. The initial framework published for comment in February 2000 represents work in progress and it is anticipated that it will change as comments are received, it is further evaluated, and the trial implementation proceeds.

The staff has held a number of public meetings with stakeholders to solicit feedback on the risk-informing process. Lines of communication have been established with industry organizations. The staff has also had several discussions with the ACRS on this topic. The staff plans to continue to interact frequently with stakeholders during the risk-informing process.

As noted above, 10 CFR 50.44 “Standards for combustible gas control system in light-water-cooled reactors” was selected for the first trial implementation of the procedures in SECY-99-264. 10 CFR 50.44 was promulgated to provide a means for the control of hydrogen gas that could be evolved following a design basis loss-of-coolant accident (LOCA) and thereby reduce the risks of a hydrogen combustion that could threaten the integrity of the containment. Further requirements were added to 10 CFR 50.44 after the TMI-2 accident to reduce the risk of hydrogen combustion from degraded core accidents in the smaller volume containments. In Phase 1 the current requirements in 10 CFR 50.44 are evaluated and potential options are identified for changes to make the regulation risk-informed. This report describes the trial implementation of risk-informing 10 CFR 50.44.

1.2 Objectives

As part of the trial implementation, 10 CFR 50.44 was selected as a “test case” for piloting the process of risk-informing 10 CFR Part 50. This study therefore has the following objective:

- To demonstrate the feasibility of the risk-informing process by applying the procedures described in SECY-99-264 [2] together with the guidance in the framework document (Attachment 1 to Reference [3]) in order to risk-inform 10 CFR 50.44. The process should therefore be able to:

- Identify and describe potential risk informed options.
- Evaluate the options with the objective of identifying potential alternatives to the current requirements in 10 CFR 50.44.
- Provide the basis for recommendations to the Commission which, if approved, would lead to initiation of rulemaking.

1.3 Scope, Limitations and General Comments

The work to determine the feasibility of risk-informed changes to the technical requirements of 10 CFR 50.44 was carried out in the following manner:

- The focus of this test case is on risk-informing 10 CFR 50.44, which deals only with the threat to containment integrity from the combustion of combustible gases generated during an accident in which significant core damage occurs. There is no intention of developing an alternative containment rule that would be capable of mitigating all potential ways of failing containment (direct containment heating, direct contact of the core debris with the containment boundary, pressure due to steam and non condensible gas generation, etc.) during all severe accidents. Thus compliance with a risk-informed 10 CFR 50.44 will only ensure that the combustion threat is mitigated.

- The intention is to provide a better balance to the 10 CFR 50.44 technical requirements among needed defense-in-depth and safety margins as well as risk. This improved balance will be achieved by systematic considerations of the requirements and may involve relaxing requirements in some areas in combination with increasing requirements in other areas.

- The study will focus on potential changes to the technical requirements associated with 10 CFR 50.44. Since the basis for these requirements may be contained in the regulations themselves or in supporting regulatory guides, standard review plan sections, branch technical positions, or other documents, all such documents are reviewed and, as necessary, considered for change.

- The study identified a requirement that, while important to safety, was found not to be directly related to the concern being addressed by 10 CFR 50.44. The requirement was retained (even though it is not directly related to the concern) rather than moving it to a more relevant rule. This was done to avoid the additional effort that would be associated with deleting the requirement and moving it to another rule.

- This test case followed the process described in SECY-99-264 [2] and the framework document (Attachment 1 to Reference [3]), and thus has the following components, which are discussed in more detail in the references:

  - The set of safety principles established in Regulatory Guide 1.174 [4] will be applied to possible changes to requirements studied in this phase.

  - The criteria applied in this case study for risk categorization will build upon and be consistent with those being used in the Option 2 work as described in SECY-99-256 [5]. It will also build upon and be coordinated with the risk-informed plant oversight process.

- The criteria established in this study with respect to needed quality of a licensee probabilistic risk assessment (PRA) will be consistent with those proposed in SECY-99-256 and RG 1.174.

- The principal focus of this work is on the current set of licensed reactors. However, potential regulatory changes that impact both current and future plants will receive higher priority than those only affecting current reactors.

1.4 Organization of Report

Chapter 2 of this report describes at a high level how the objective of risk-informing the regulations in Part 50 will be accomplished and how success will be measured. The approach follows the proposed framework for the risk-informing process described in Attachment 1 to Reference [3]. This and subsequent chapters in the report describe how this approach is applied to the requirements in 10 CFR 50.44.

A detailed examination of 10 CFR 50.44 was performed and is described in Chapter 3. Initially the analytical and physical requirements actually imposed by 10 CFR 50.44 are identified and described. Any relationship of 10 CFR 50.44 to other regulations and implementing documents is then identified. This information is needed because changes to 10 CFR 50.44 could potentially impact some of the related regulations or the implementing documents. It is also necessary to understand how the requirements in 10 CFR 50.44 are actually implemented both from the industry and the regulators perspective. This chapter is therefore intended to provide a clear picture of the requirements in 10 CFR 50.44 (and the supporting documentation) and indicate how they are implemented.

In Chapter 4 the concern (i.e., the threat to containment integrity from combustion) associated with 10 CFR 50.44 is described. The risk significance of combustion for the various containment designs and the associated needs for control of combustible gases are determined and also discussed. The purpose of this chapter is to identify the needed attributes for a risk-informed 10 CFR 50.44. This information taken together with the information in Chapter 4 is used to identify risk-informed options in Chapter 5.

Two approaches are used in Chapter 5 to develop potential risk-informed options for 10 CFR 50.44. One approach develops options based on the existing requirements in 10 CFR 50.44. The other approach addresses the concern (to be dealt with by 10 CFR 50.44) by systematically applying the strategies developed in the framework (Attachment 1 to Reference [3]) for risk-informed changes to 10 CFR Part 50. Both approaches have the same overall objective which is to develop risk-informed options dealing with the identified concern.

Finally, in Chapter 6, the implications of the alternative are evaluated and a preliminary assessment is presented in the report.

1.5 References

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

2. USNRC, "Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR PART 50," SECY-99-264, November 8,1999.

3. USNRC, "Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3)," SECY-00-0086, April 12, 2000.

4. USNRC, "An Approach for Using Probabilistic Risk Assessments in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.

5. USNRC, "Rulemaking Plan for Risk-Informing Special Treatment Requirements," SECY-99-256, October 29, 1999.

2. APPROACH

This section describes at a high level how the objective of identifying risk-informed changes to the technical requirements of 10 CFR Part 50 will be accomplished and success measured. This section summarizes a framework (Attachment 1 to Reference [1]) developed by the staff to more clearly define and guide the process for identifying risk-informed changes. A representation of this framework is presented in Figure 2.1.

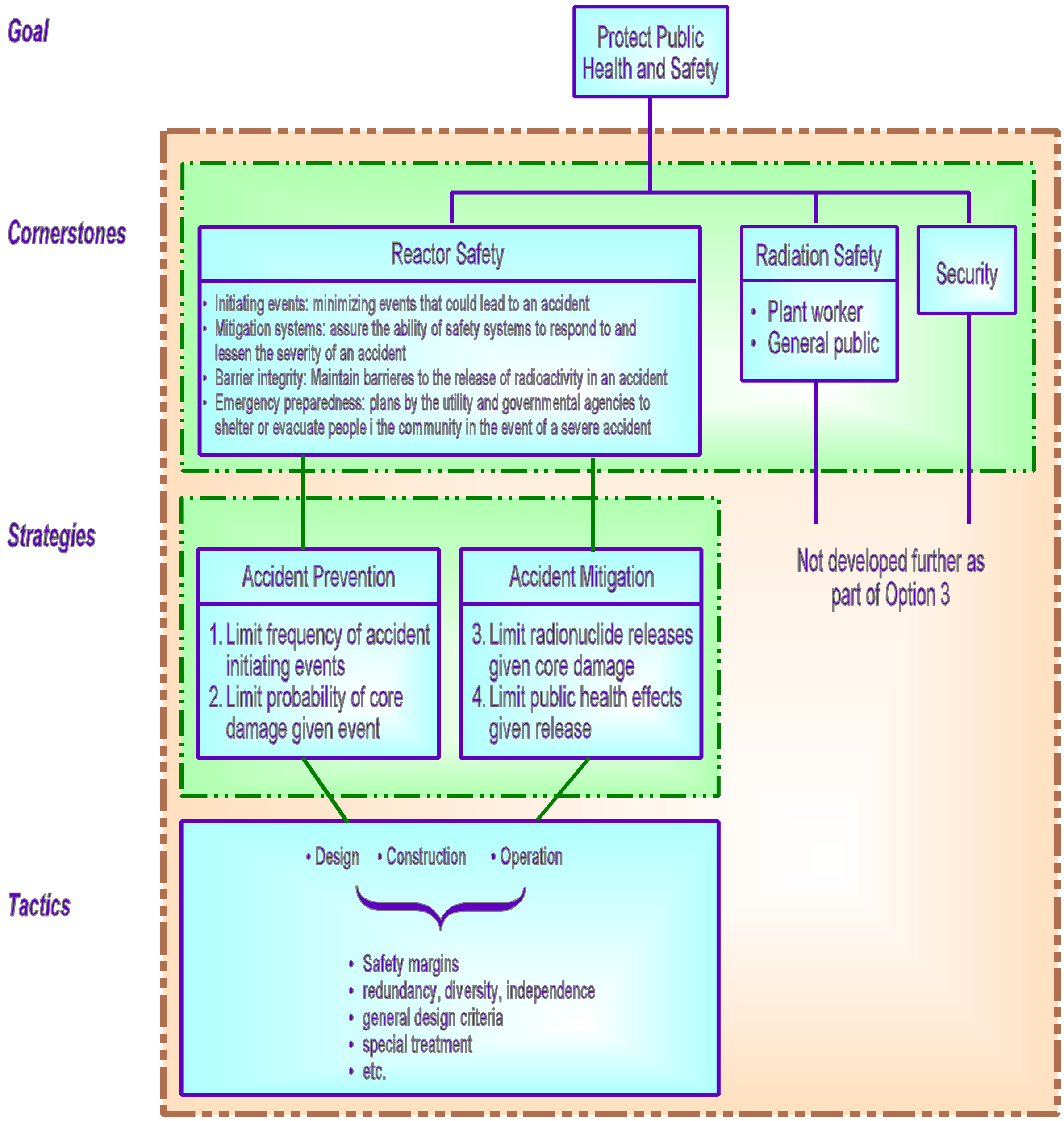

Figure 2.1 Risk-Informed Defense-in-Depth Framework

The structure and elements of the above framework are consistent with established regulatory philosophy and have as a high level goal the protection of the public health and safety. A balanced high-level defense-in-depth approach (based on prevention and mitigation) is included in the framework to help achieve this goal. The approach is summarized in the following working definition:

Defense-in-depth is the approach taken to protect the public by applying the following strategies in a risk-informed manner:

(1) limit the frequency of accident initiating events (initiators)

(2) limit the probability of core damage given accident initiation

(3) limit radionuclide releases during core damage accidents

(4) limit public health effects due to core damage accident

The strategies are applied considering the following defense-in-depth elements:

- *reasonable balance is provided among the strategies*

- *over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided*

- *independence of barriers is not degraded*

- *safety function success probabilities commensurate with accident frequencies, consequences, and uncertainties are achieved via appropriate*
  - *redundancy, independence, and diversity,*
  - *defenses against common cause failure mechanisms,*
  - *defenses against human errors, and*
  - *safety margins*

- *the defense-in-depth objectives of the current General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50 are maintained*

Various tactics are used to support the high-level strategies, and existing regulatory requirements deal with implementation of such tactics. The above framework is extended to include quantitative guidelines for risk-informing existing technical requirements. The intent is to develop risk-informed regulations, which retain deterministic characteristics, in such a way that compliance provides reasonable assurance of protection of the public health and safety. The quantitative guidelines for the different strategies are provided in Figure 2.2. These values are consistent with the quantitative health objectives (QHOs), and the subsidiary objectives in current use. The quantitative guidelines in Figure 2.2 are used in the implementation process to determine potential risk-informed options for each of the above strategies for a given regulatory requirement.

	Accident Prevention		Accident Mitigation	
	Core Damage Frequency $\leq 10^{-4}$/year		Conditional Large Early Release Probability $\leq 10^{-1}$ (Note 5)	
	Limit frequency of accident initiating events	Limit probability of core damage given event	Limit radionuclide releases given core damage	Limit public health effects given release
	Initiator Frequency	Conditional core damage probability	Conditional large early release probability	Conditional individual fatality probability
Frequent initiators	≥ 1/year	$\leq 10^{-4}$	$\leq 10^{-1}$	Note 3
Infrequent initiators	$\leq 10^{-2}$/year	$\leq 10^{-2}$	$\leq 10^{-1}$	Note 3
Rare initiators	$\leq 10^{-5}$/year	Note 4	Note 4	Note 3

Notes:
1. The product across each row gives a large early release frequency of $<10^{-5}$/year.
2. It is preferable that no single type of initiator cause a large fraction of any frequency guideline.
3. No quantitative guideline is proposed for the fourth strategy, the LERF guideline is used as a surrogate.
4. For rare initiators, emphasis is placed on Strategy 1, limit initiator frequency.
5. Measures to mitigate late large releases are also appropriate. A conditional probability of a late large release (up to 24 hours after the onset of core damage) of $\leq 10^{-1}$ is proposed.

Figure 2.2 Quantitative Guidelines for Risk-Informing Regulations

The definition of LERF in the above figure is the same as in Regulatory Guide 1.174. LERF is the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines.

Not every containment bypass or early failure would result in a large release. To be risk-significant containment leakage must far exceed the design basis containment leak rate. Containment failure modes that result in scrubbed releases or leak paths that are isolated before the onset of significant core damage generally do not lead to large releases.

For some plants, large releases could occur hours after reactor vessel bottom head failure. An example would be a release due to containment overpressurization or high temperature while core-concrete interactions are proceeding in the absence of an overlying water pool. Containment heat removal systems may be inoperable in this scenario, and natural processes would take hours after the completion of core-concrete interactions to remove radionuclides from the containment atmosphere.

Effective evacuation can mitigate the threat of acute health effects offsite given such a delayed large release. However, there are accidents in which external events may preclude or hinder evacuation efforts. Plant workers would also need to be protected from any delayed large release. A quantitative guideline was therefore included in Figure 2.2 to reflect the need for defense-in-depth against the threats posed by such delayed releases. Specifically, the conditional probability of a large late release should be 10^{-1} or less. Late in this context extends to approximately 24 hours after the onset of core damage.

In implementing the framework (with its quantitative guidelines), three major steps are followed as depicted in Figure 2.3: (1) the selection of the regulation to be risk-informed, (2) the development of the risk-informed alternative, and (3) the evaluation of the risk-informed alternative.

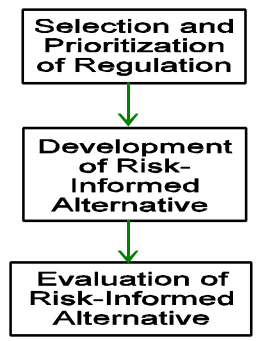

Figure 2.3 Process for Risk-Informing Regulations

The entire process dictated by this approach is likely to be highly interactive as more experience is gained in the development and evaluation of risk-informed options.

2.1 Selection of Regulation

The first major element in the process is the selection of the regulation that needs to be risk-informed. The selection process consists of five major components as shown in Figure 2.4.

A coarse screening of the regulations in Part 50 is initially performed to determine whether the regulation has an impact on prevention or mitigation of core-damage accidents, because these present the most risk to the public and risk information is most prevalent for such accidents. Only those regulations that have potential relevance to safe plant design, operation, or maintenance are candidates for risk-informing. A regulation may become a candidate for elimination if it does not impact any of the strategies or has an insignificant impact on the quantitative guidelines embedded in the three strategies delineated in Figure 2.2. As the second step in Figure 2.4 indicates, as part of this coarse screening an attempt is also made to identify risk-significant safety issues not implicitly addressed in the current regulations. Another screening is then performed, as the third step, to identify those regulations that do not warrant risk-informing and can be eliminated from further consideration; i.e., there is no need for safety improvement, there is no excess conservatism or margin in the regulation, and there is no unnecessary burden created by excessive conservatism. Fourth, an evaluation of the regulations is performed to identify any ties, overlaps or redundancies among regulations to determine if any should be "linked/grouped" as a "single" risk-informed regulation. Finally, the remaining set of regulations is prioritized.

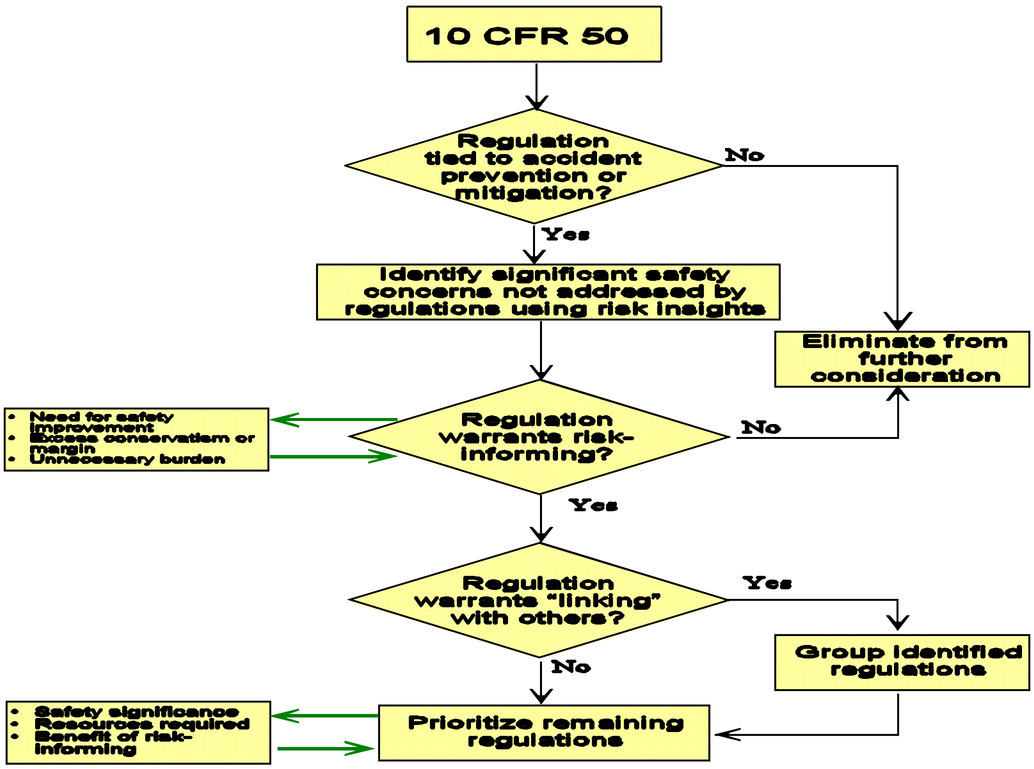

Figure 2.4 Process for selection and prioritization of regulations to be risk-informed

The prioritization considers three factors:

- the safety significance of the regulation
- the potential resources required to risk-inform (considering complexity, information requirements, need for a demonstration plant, time, manpower, etc.), and
- the benefit of making risk-informed changes to the regulation (e.g., the potential for reducing unnecessary burden).

2.2 Development of Risk-Informed Changes

The second major element in the process is to develop the changes to the technical requirements for the high-priority regulations identified in the first element of the process. Two approaches are followed for developing risk-informed changes to a regulation. Both approaches begin with an examination of the concern or concerns that necessitated the regulation, and both approaches have the same overall objective, which is to develop risk-informed requirements for dealing with the identified concern. However, one approach starts from the current set of technical requirements of the regulation and attempts to develop risk-informed changes by analyzing the technical requirements. The second approach takes a fresh start by applying the four high-level defense-in-depth strategies; in effect, ignoring the existing technical requirements of the regulation.

There are two principal reasons for following two approaches to developing a risk-informed alternative to a regulation. The first reason is for completeness. Following both of the above approaches gives greater confidence that all reasonable risk-informed options have been identified. The second reason is to identify a risk-informed alternative that is the most optimal by looking at the concern from an alternative perspective, that is, without being constrained, or unduly influenced, by the existing requirements.

Potential changes identified by either of these two approaches are developed based on the following six considerations:

- risk insights from plant specific PRAs
- industry experience
- consistency with the quantitative guidelines identified in the framework document
- reasonable cost burden
- proven technology
- suitability for performance-based compliance monitoring

The potential changes derived from both approaches are evaluated to arrive at the risk-informed alternative.

2.2.1 Revising Current Requirements Approach

The approach based on revising the existing technical requirements is shown in Figure 2-5. Each of the six steps in this approach is described below.

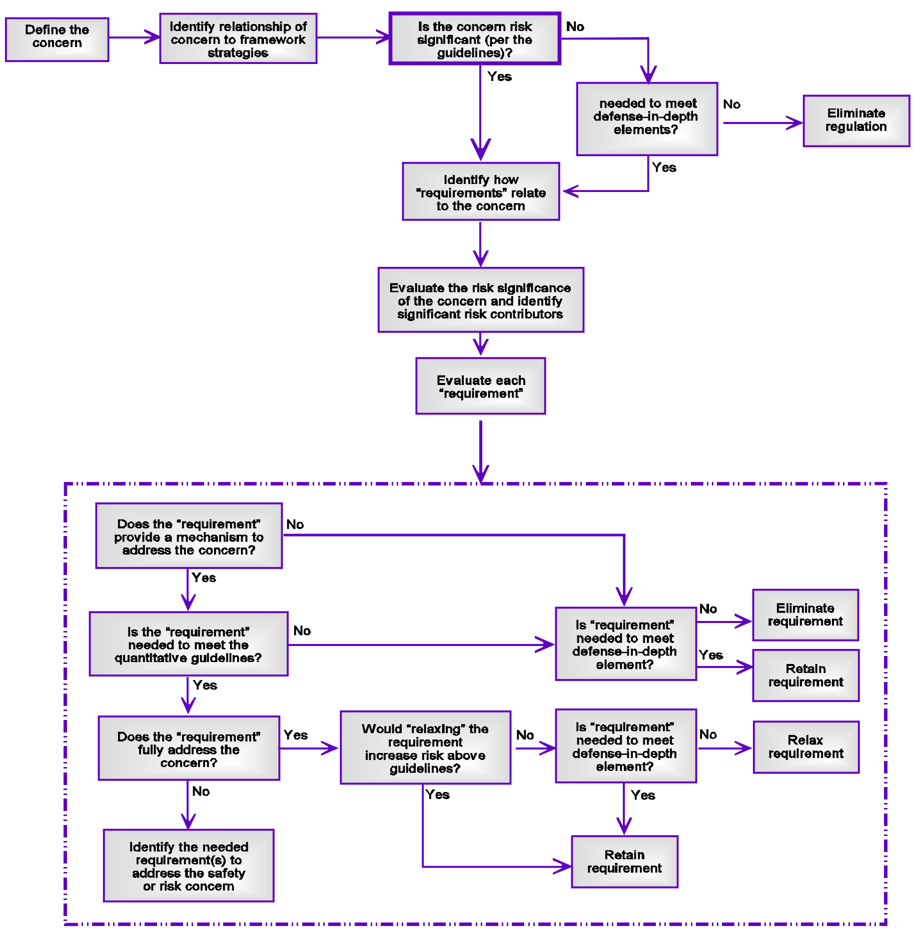

Figure 2.5 Current Requirements Approach to Develop Risk-Informed Options

(1) Define the concern:

As mentioned previously, development of risk-informed changes to the technical requirements of a regulation begins with an examination of the concern or concerns that necessitated the regulation. Only after the concern is clearly understood, can a determination be made as to how risk-significant the concern is, and how effectively the concern is addressed by the existing requirements. The concern should be expressed in terms of its risk significance (e.g., which risk-significant accidents are impacted, and how significant is this impact).

(2) Identify relationship of concern to framework strategies:

In Section 2.1 of this report, the four defense-in-depth strategies to be considered in making risk-informed changes to the regulations were identified. Two of these strategies are preventive (limit frequency of accident initiators and limit probability of core damage given an initiator), and two of the strategies are mitigative (limit radionuclide releases given core damage and limit public health effects given release). The next step in developing risk-informed changes is to identify which of the four strategies are impacted by the concern.

(3) Is the concern risk significant (per the guidelines)?

The risk significance of the concern is assessed against the quantitative guidelines in Figure 2-2. Based on information derived from PRAs, an assessment of the quantitative significance of the concern is made with respect to the quantitative guidelines presented in Figure 2-2 for various types of plants (as defined by their nuclear steam supply systems or containment designs). If the risk significance of the concern results in values significantly below the quantitative guidelines, then the regulation (in its entirety) may become a candidate for elimination. Such regulations must be evaluated to determine (1) if the low risk is because of the technical requirements imposed by the regulation, and if not, then (2) whether the technical requirements are needed to meet any of the defense-in-depth elements. If it is determined that they are not needed to meet the guidelines nor are they needed to maintain defense-in-depth, then the regulation itself becomes a candidate for elimination.

(4) Identify how "requirements" relate to the concern:

Each technical requirement contained in the existing regulation is identified and described in detail in terms of the affected systems, structures, components and procedures (if any) for the various types of plants and the criteria used for assessing compliance with the requirements. A review is then made to determine the relationship of each requirement to other regulations and implementing documents, such as regulatory guides, standard review plan, branch technical positions, generic letters, etc. The purpose of this review is to obtain a detailed understanding of the implications of revising any particular requirement in terms of its impact across the body of the regulations and implementing documents.

Subsequent to the above review, the basis and method of implementation of the requirements by industry are identified and described. A determination is made as to whether the requirement has been implemented by the licensees on the basis of the regulation alone, on the basis of an associated regulatory guide or other implementing document, or on some other basis.

Lastly, each requirement identified at the beginning of this step is evaluated in the context of how effectively it addresses the defined concern.

(5) Evaluate the risk significance of the concern and identify significant risk contributors:

This step is essentially a detailed extension of step (3), above. In step (3), the risk significance of the concern was evaluated, at a high level, in comparison with the quantitative guidelines provided in Figure 2-2. Given that the concern was determined to be risk-significant in step (3), in this step, available PRA information (e.g., NUREG-1150, or IPEs) is reviewed to determine what is driving the risk-significance of the concern. For the various types of plants (as defined by their nuclear steam supply systems or containment designs), the risk significant contributors are identified, where possible, in terms of the PRA results (e.g., dominant accident sequences, or dominant containment failure modes).

(6) Evaluate the each requirement:

In this step, each technical requirement identified in step (4) is evaluated to determine if, and how, it should be risk-informed. Options for risk-informing the technical requirements broadly fall into the following three categories:

- eliminate the current requirement
- retain the current requirement
- revise, enhance, or supplant the current requirement

Guidance as to which category each requirement falls into is provided by answering the three questions described below.

1. Does the requirement provide a mechanism to address the concern?

The answer to this question should have been obtained during step (4) above. If the requirement does not provide a mechanism to address the concern, then it should be evaluated to determine whether it is needed to meet any of the defense-in-depth elements. If it is determined that the requirement is needed to maintain defense-in-depth, then the requirement is retained. However, if it is determined that the requirement is not needed to maintain defense-in-depth, then it becomes a candidate for elimination. It is important to note, as before, that all requirements that are identified as candidates for elimination through this process will also be examined to assure that their elimination will not have any adverse impact on the radiation safety and security cornerstones.

If the requirement does provide a mechanism to address the concern, then it is subjected to the following question.

2. Is the requirement needed to meet the quantitative guidelines?

Based on information obtained in steps (3-5), a determination is made as to whether the requirement is necessary in order for the strategies impacted by the concern to meet the associated quantitative guidelines provided in Figure 2-2. If the requirement is determined to not be necessary to meet the quantitative guidelines, then it will be either eliminated or retained based on whether it is needed to meet any of the defense-in-depth elements, as discussed for the previous question. If the requirement is determined to be necessary to meet the quantitative guidelines, then it is subjected to the following question.

3. Does the requirement fully address the concern?

This question is used to determine whether or not a safety enhancement would be appropriate. It is possible that there are aspects of the defined concern which are not fully addressed by the existing requirement (or requirements). In this case, any necessary additional requirements should be identified, so that the concern will be fully addressed.

If the requirement does fully address the concern, then the requirement should be evaluated to determine whether or not it can be relaxed and still maintain risk below the quantitative guidelines.

If relaxing the requirement would increase risk above the guidelines, then the requirement is retained, as is. If relaxing the requirement would still maintain risk below the guidelines, then it can be relaxed, as long as it is not needed to meet any of the defense-in-depth elements, as discussed previously.

2.2.2 Developing Alternative Requirements Approach

As noted above the main difference between the two approaches to developing risk-informed changes is that risk-informed changes obtained through implementation of the alternative requirements approach are developed without reference to the existing technical requirements of the regulation. In this approach, as seen from Figure 2-6, risk-informed changes for addressing the concern can be identified during any of steps (2-4) of the figure. This allows changes to be developed from different perspectives. The four steps in an alternative approach that begins afresh from a risk-informed perspective are described below using the four strategies of the framework and again defining the concern.

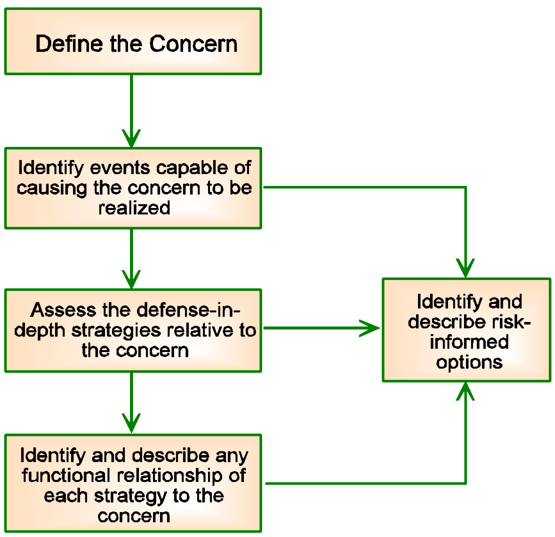

Figure 2.6 Alternative requirements approach to developing risk-informed options

(1) Define the concern:

This step is very similar to step (1) for the revising current requirements approach. As mentioned previously, development of risk-informed changes to a regulation begins with an examination of the concern or concerns that necessitated the regulation. The concern should be expressed in terms of its risk significance (e.g., which risk-significant accidents are impacted, and how significant is this impact).

(2) Identify events capable of causing the concern to be realized:

After the concern is defined, an identification is made at a high-level of events that could cause the concern to be realized. For example, if the concern is that a deflagration/detonation of combustible gas could threaten containment, for the concern to be realized there must be generation of combustible gas from metal-water reactions during an accident in which significant core damage occurs. Existing PRAs can, generally, provide more specific insights regarding specific sequences of events that are most likely to cause an identified concern to be realized.

(3) Assess the defense-in-depth strategies relative to the concern:

As mentioned previously, Section 2.1 of this report identifies four defense-in-depth strategies for limiting accident risk. Three of these strategies also have quantitative guidelines associated with them, as shown in Figure 2-2. In this step, the efficacy of each strategy relative to preventing and mitigating the identified concern is assessed. For those strategies that address the concern, performance-based options can be developed with high-level acceptance criteria, which would allow licensees substantial flexibility in meeting them. In addition, if it is anticipated that it may be difficult for licensees to meet the high-level acceptance criteria based on the strategies that address the concern, similar type options can be developed based on the remaining strategies. For example, the reduction of the frequency of an accident class under which the concern becomes less manageable may be more practical than ensuring the operability of a mitigating system under the same conditions.

(4) Identify and describe any functional relationship of each strategy to the concern:

Understanding the functional relationships between each strategy and the concern allows practical methods of applying each defense-in-depth strategy to the defined concern to be identified, for relevant plant types. These changes are expected to be much more prescriptive than those developed under the preceding step. For example, specific hardware or procedures may be identified in these changes for applying a specific strategy to the concern. As in the previous step, the changes may relate to the strategies that address the concern, or it may prove to be more practical to develop changes related to the other strategies. For example, station blackout accidents may impose the most severe conditions on the plant's ability to successfully control combustible gas concentrations. An option reducing the frequency of station blackout may prove to be more practical for managing the defined concern than attempting to ensure that mitigating systems can successfully operate under station blackout (SBO) conditions.

2.3 Evaluation of Risk-informed Alternative

In the previous step, all changes were developed based on safety and risk implications. These changes were evaluated to arrive at a risk-informed alternative to an existing regulation. In this step, the risk-informed alternative is evaluated in order to estimate the associated NRC and licensee burdens, for both implementing and applying the alternative, and to compare these estimates with similar estimates for the existing regulation. The factors affecting both NRC and licensee burden are listed below.

Factors impacting NRC:

- Need for a rule change
- Impact on other regulations
- Need to revise or modify implementing documents
- Need to create a new implementing document
- Extent of regulatory analysis required
- Need for NRC review of licensee submittals
- Impact on NRC inspection activities

Factors impacting Licensees:

- Need for new or modified equipment
- Need for analysis
- Impact on maintenance and inspection activities
- Impact on technical specifications
- Impact on procedures and training

2.4 References

1. USNRC, “Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3),” SECY-00-0086, April 12, 2000.

3. EXAMINATION OF 10 CFR 50.44

3.1 Selection of Regulation

SECY-99-264, “Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR 50" [1] provides the staff's plan for the study phase of its work to risk-inform the technical requirements of 10 CFR Part 50 (i.e., Option 3 of SECY-98-300 [2]). SECY-99-264 notes that the staff intends to test the process outlined in the plan by using at least two example 10 CFR Part 50 modifications, one of which involves the modification of a single requirement (e.g., hydrogen control requirements in 10 CFR 50.44) and one which involves modification of a set of related requirements (e.g., requirements related to special treatment of SSCs).

The selection of 10 CFR 50.44 as a test application has been prompted in part by the fact that a number of licensees have identified 10 CFR 50.44 as a regulation which includes requirements that may not be risk significant, and whose implementation therefore places unnecessary burden on the licensees. In a Nuclear Energy Institute (NEI) survey [3] on the need and benefit of improving NRC Technical Requirements, 56 plants responded, and 24 units identified 10 CFR 50.44 as a high priority candidate for change. 10 CFR 50.44 also was the subject of an exemption request [4] from a licensee, Southern California Edison, that operates the San Onofre Nuclear Generating Station (SONGS), a pressurized water reactor (PWR) with a large dry containment. Specifically, SONGS requested an exemption from 10 CFR 50.44 to remove requirements for hydrogen control systems in accordance with the pilot program for risk-informed, performance-based regulation. The petition was granted by the NRC and the staff recognized that the basis for the exemption was not SONGS specific, but was applicable on a wider, generic basis. In accordance with NRC Commission guidance, rulemaking should be used to avoid numerous exemption requests. Subsequent to the San Onofre exemption, the NRC staff received additional requests for relaxation of some regulatory requirements found in 10 CFR 50.44, specifically from the Boiling Water Reactor (BWR) Owners Group.

More industry support for 10 CFR 50.44 as a candidate to be risk informed was shown at the NRC-Sponsored Public Workshop on Options for Risk-Informed Revisions to Part 50 [5] held on September 15, 1999 in Rockville, MD. At the workshop, an Open Discussion session focused on the identification of candidate requirements and design basis accidents to be revised, particularly on the selection of top candidate(s) for risk-informing. Several stakeholders expressed views that 10 CFR 50.44 should be such a candidate for risk-informing on the basis that some of the requirements of the current regulation do not contribute to risk reduction and cause unnecessary burden.

10 CFR 50.44 also becomes a viable candidate for risk-informing when the selection criteria described in Section 2.1 are applied:

- 10 CFR 50.44 affects accident prevention and mitigation.
- 10 CFR 50.44 warrants risk-informing it may not address the safety issue of concern most efficiently or effectively, and may impose excess burden.
- 10 CFR 50.44 is directly linked with other regulations (e.g., 10 CFR 50.47).

The ability to control combustible gases is directly tied to the defense-in-depth concept of accident prevention and mitigation. Specifically, hydrogen combustion maybe a direct threat to the containment integrity (i.e., ability to mitigate an accident), but could also be minimized by preventive strategies.

10 CFR 50.44 warrants risk-informing since, when examining current requirements, certain parts of 10 CFR 50.44 appear to be designed to mitigate accidents that are not risk significant, and consequently, appear to impose unnecessary burden. For instance, it is likely that the removal of some aspects of the hydrogen control systems for LOCA, or a reduction of their surveillance and maintenance requirements would be cost beneficial. In addition, there appear to be risk significant accidents that the current requirement do not explicitly address.

10 CFR 50.44 is related to other regulations, for example, 10 CFR 50.47, Emergency Plans, which requires that the *"emergency preparedness provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency."* Further, 50.47 requires that this emergency plan meet the requirement of Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities. Appendix E, Section VI, Emergency Response Data System, requires the licensee to provide data on selected plant parameters, one being hydrogen concentration. In risk-informing 10 CFR 50.44, the impact to the other regulations also needs to be addressed.

3.2 Description of Regulation

The present form of section 44 of part 50 of title 10 of the Code of Federal Regulations (10 CFR 50.44), "Standards for combustible gas control system in light-water-cooled power reactors," results from the original rule of 1978 and two major amendments motivated by the 1979 accident at Three Mile Island Unit 2. One amendment was incorporated into the rule in 1981, the other in 1985. The various requirements are described below in terms of the original rule and its amendments.

3.2.1 Original Rule

Because of the potential for hydrogen generation as a result of a LOCA, the NRC published, on October 21,1976, in the Federal Register (41FR 46167) a notice of proposed rulemaking concerning proposed amendments to 10 CFR Part 50. The proposed rule was published in the Federal Register (43 FR 50162) and became part of the Code of Federal Regulations, as 10 CFR 50.44, in 1978.

The logic of the original rule is illustrated in Figure 3.1 below. The letters below the boxes indicate the part of the rule referred to.

10 CFR 50.44 requires each light-water-cooled reactor (LWR) fueled with oxide pellets encased within zircaloy cladding to have a means for controlling hydrogen gas generated following a postulated LOCA. The hydrogen gas could be generated by: (1) metal-water reaction between the zirconium cladding and the reactor coolant, (2) radiolytic decomposition of the coolant, and (3) corrosion of metals.

In controlling the generated gas, each boiling or pressurized LWR is required to have a capability for:

- measuring the concentration of hydrogen in the containment
- insuring a mixed atmosphere in the containment, and
- controlling combustible gas concentrations in containment following a postulated LOCA.

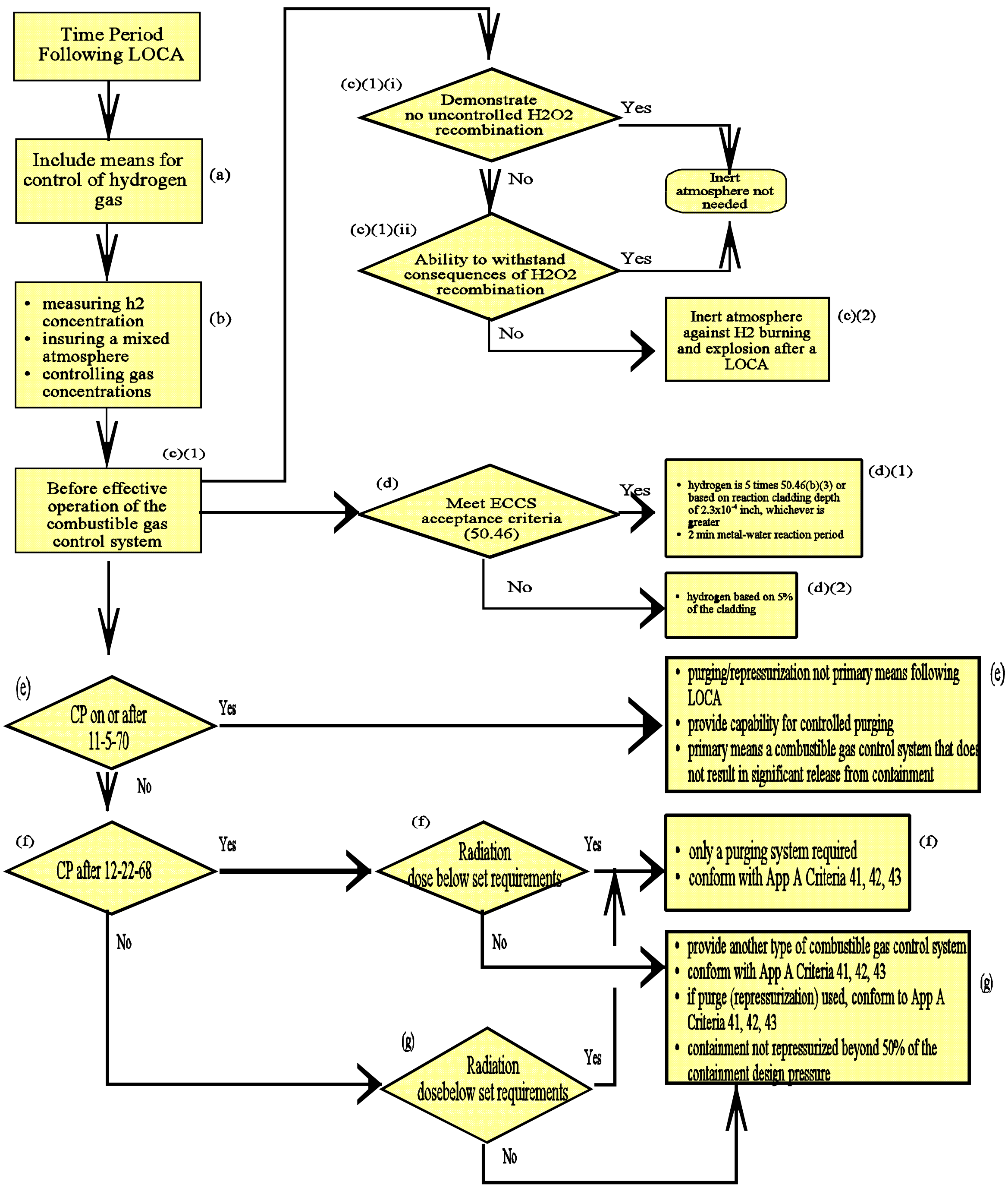

Figure 3.1 Requirements of the Original 10 CFR 50.44 Rule

In addition, for each BWR or PWR, it must be shown, during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, that either: (1) an uncontrolled H2-O2 recombination would not occur within the containment, or (2) the plant could withstand the consequences of an uncontrolled recombination without loss of safety function. If these two conditions cannot be demonstrated then the containment shall be provided with an inerted (oxygen deficient) atmosphere to provide protection against hydrogen burning and explosion (e.g., deflagration or detonation) during the time period specified above.

Regarding the amount of hydrogen to be considered for the combustible gas control system, the rule stated the following:

- For facilities that are in compliance with 50.46(b), i.e., the acceptance criteria for emergency core cooling systems, specifically, the peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling, the amount of hydrogen due to core metal-water reaction (% of cladding that reacts with water) shall be assumed to be either five times the total hydrogen calculated in demonstrating compliance with 50.46 (b) (3) or the amount that would result from reaction of all the metal on the outside of the cladding on the rods to a depth of 0.00023 inch, whichever is greater. (In calculating the hydrogen generated, 50.46 (b) (3) calls for the assumption of a maximum of 1% clad metal-water reaction). A time period of 2 minutes shall be used as the time interval following the postulated LOCA over which the metal-water reaction occurs.

- For facilities which have no evaluation of compliance with 50.46 (b), the amount of hydrogen generated shall be assumed to be equivalent to that occurring from a 5% clad metal-water reaction

Regarding the type of combustible gas control systems which would be acceptable, the rule stated the following:

- For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970, purging and/or repressurization shall not be the primary means for controlling combustible gases following a LOCA. However, the capability for controlled purging shall be provided. For these facilities, the primary means for controlling combustible gases following a LOCA shall consist of a combustible gas control system, such as recombiners, that does not result in a significant release from containment.

- For facilities with respect to which the notice of hearing on the application for a construction permit was published prior to November 5, 1970, if the incremental radiation dose from purging (and repressurization if a repressurization system is provided) at all points beyond the exclusion area boundary after a postulated LOCA ...*(is within certain limits)...* and if the combined radiation dose at the LPZ outer boundary from purging and the postulated LOCA....*(is within certain limits)*....., only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to part 50. Otherwise the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to part 50. If a purge system is used as part of the repressurization system, the purge system shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of appendix A to part 50. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

***In summary**, the requirements imposed by the original rule of 10 CFR 50.44 include the following:*

Analytical requirements

- *the type of accident considered, viz. postulated LOCA*
- *the sources of hydrogen (fuel-cladding oxidation, radiolysis, and corrosion)*
- *the hydrogen source term (5% oxidation reaction over a 2 minute period)*

Physical requirements

- *measuring hydrogen concentrations*
- *insuring a mixed containment atmosphere*
- *controlling combustible gas concentrations resulting from a postulated LOCA*

3.2.2 Amendments to 10 CFR 50.44

In the aftermath of the TMI accident, the NRC reevaluated the adequacy of the regulations related to H2 control with the intent of providing greater protection in the event of accidents more serious than design basis LOCAs. Specifically, significant quantities of hydrogen from the metal-water reaction, estimated at approximately 400 kg, were generated during the core melt accident at TMI-2 on March 28, 1979. Combustion of the hydrogen released to containment during the accident sequence generated a pressure spike of about 28 psig (peak pressure). Since the design pressure of the large dry containment at TMI-2 was approximately 60 psig, the accident pressure spike did not pose a threat to containment integrity. However, the occurrence of the extensive metal water reaction and subsequent hydrogen burn in the TMI-2 accident gave impetus to the imposition of additional hydrogen control requirements that included additional hardware backfits to the small volume pressure suppression containments such as the BWRs and the ice condenser PWRs.

In addition, during the TMI accident a hydrogen “bubble” was formed in the reactor coolant system, which impeded adequate coolant flow to the reactor core. As a result new requirements were also imposed as part of 10 CFR 50.44 that required installation of high point vents in the RCS of all plants to allow venting of non-condensible gases.

3.2.2.1 1981 Amendment

In 1981, the NRC published (46FR58484) amendments to 10 CFR 50.44, “Interim Requirements Related to Hydrogen Control.” This amendment added Sections (c)(3)(i), (c)(3)(ii), and (c)(3)(iii) to the rule.

As illustrated in Figure 3.2, the 1981 amendment imposed three requirements which included:

- Inerted atmosphere for Mark I and Mark II containments

- Installation of recombiners for LWRs that rely on a purge or repressurization system as a primary means of controlling combustible gases following a LOCA

- Installation of high point vents.

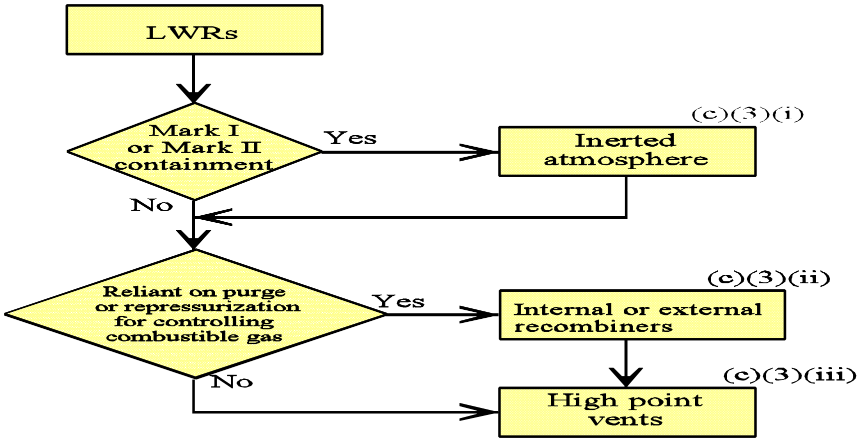

Figure 3.2 Requirements of the 1981 Amendment to 10 CFR 50.44

Regarding BWR plants with Mark I or Mark II containments, Section (c)(3)(i) of the amendment unequivocally stated that: Effective May 4, 1982 or 6 months after initial criticality, whichever is later, an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment

Also, each light-water nuclear reactor that relies upon a purge/repressurization system to control combustible gases following a LOCA is required under Section (c)(3)(ii) of the 1981 amendment to 10 CFR 50.44 to be provided with either internal or external recombiners. Whether or not internal or external recombiners are used, they must all meet the combustible gas control requirements. This amendment was subsequently modified by Generic Letter GL 84-09 [6], which exempted plants with a Mark I containment from the amendment. Section (c)(3)(ii) is illustrated in Figure 3.3 below.

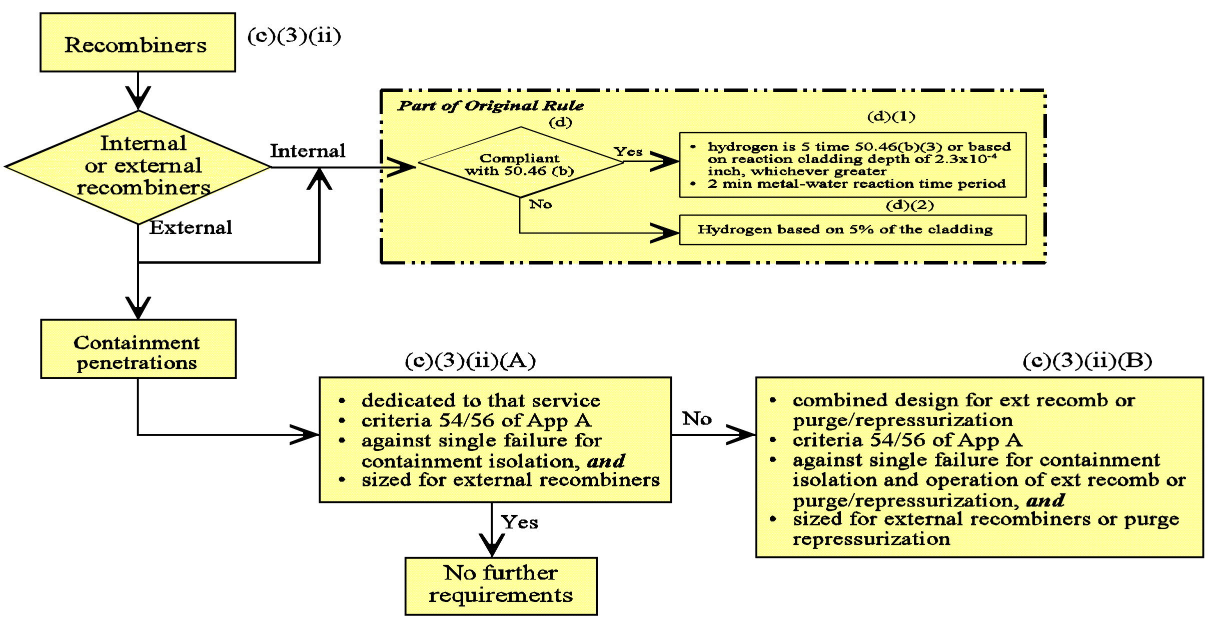

Figure 3.3 Requirements of the 1981 Amendment to 10 CFR 50.44 (Recombiners)

A large fraction of the hydrogen generated during the TMI accident accumulated in the upper region of the reactor vessel head. As this non-condensible gas "bubble" could not be vented it stagnated flow to the core and caused inadequate core cooling. In response to this problem each light-water nuclear power reactor was required, in Section (c)(3)(iii) of the 1981 amendment to 10 CFR 50.44, to be provided with high point vents for the reactor coolant system, the reactor vessel head, and for other systems. High point vents were, however, not required for tubes in u-tube steam generators. The requirements of Section (c)(3)(iii) are listed in Figure 3.4.

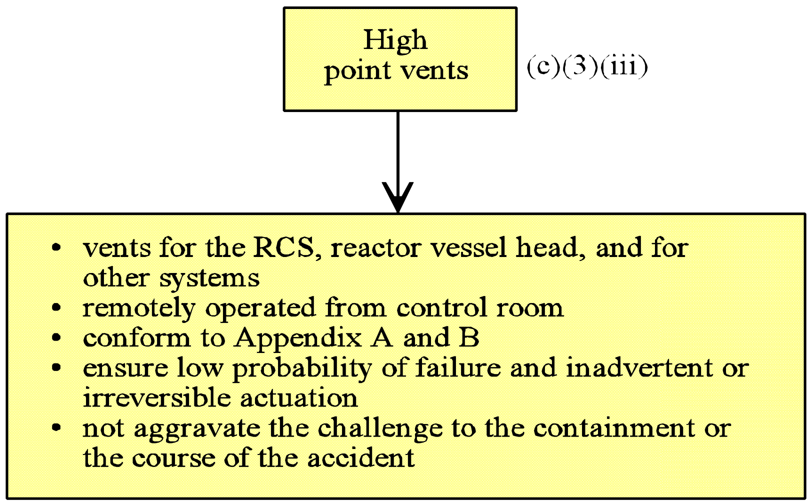

Figure 3.4 Requirements of the 1981 Amendment to 10 CFR 50.44 (High Point Vents)

***In summary**, the requirements imposed by the 1981 amendment to 10 CFR 50.44 include the following:*

- *inert Mark I and II containments*
- *recombiners for post LOCA*
- *high point vents*

3.2.2.2 1985 Amendment

In 1985, the NRC published (50FR3498) another amendment to 10 CFR 50.44, "Hydrogen Control Requirements," contained in Section (c)(3)(iv).

The 1985 amendment required a hydrogen control system for BWRs with Mark III containments and PWRs with ice condenser containments justified by a suitable program of experiment and analysis. Mark III and ice condenser plants that do not rely on inerting must have systems and components to establish and maintain safe shutdown and containment integrity and these systems must be able to function in an environment after burning and, possibly, detonation of hydrogen unless it is shown that such events are unlikely to occur. The amount of hydrogen to be considered is that generated from an equivalent 75% metal-water reaction.

Figure 3.5 shows the requirements of 10 CFR 50.44 from the 1985 amendment.

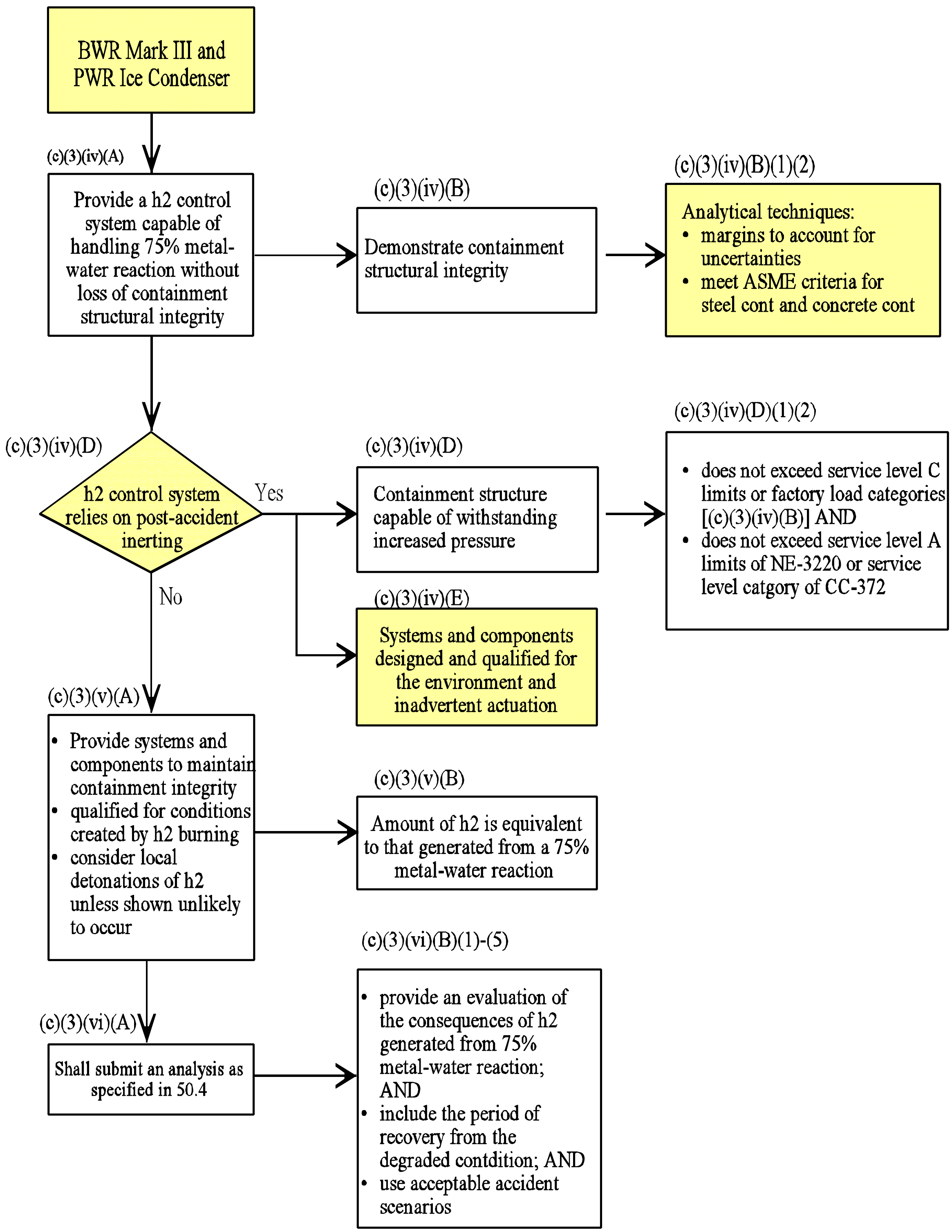

Figure 3.5 Requirements of 1985 Amendment to 10 CFR 50.44

Containment structural integrity must be demonstrated using an analytical technique acceptable to the NRC staff. An acceptable method could include the use of actual material properties with suitable margins to account for uncertainties or alternatively follow specific criteria of the ASME Boiler and Pressure Vessel Code.

If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the additional pressure either during the accident (i.e., demonstrate that Service Level C limits are not exceeded) or in the event of an inadvertent full inerting during normal plant operation (i.e., demonstrate that Service Level A limits are not exceeded). The systems required to establish and maintain safe shutdown must be qualified for the environment caused by such inerting. Inadvertent inerting during normal operation must not adversely affect systems and components needed for safe plant operation.

The analysis that Mark III and ice condenser plants are required to submit must be such that it (a) provides an evaluation of the consequences of the large amount of hydrogen (i.e., 75% metal-water reaction) assumed to be generated, including consideration of hydrogen control measures, (b) includes the period of recovery from degraded conditions, (c) uses accident scenarios accepted by the NRC staff, (d) supports design of the hydrogen control system, (e) shows that for those reactors that do not rely upon inerting to control hydrogen, the structural integrity of the containment will be maintained and the systems and components necessary to establish and maintain safe shutdown will be capable of performing their functions in the environment prevailing after hydrogen combustion and, possibly, local detonations (unless it can be shown that these events are unlikely to occur).

As originally proposed, the new requirements were applicable to PWRs with large dry containment. However, the NRC agreed with comments suggesting that implementation for these containments be deferred pending completion of severe accident rule making, at which time the results of research and PRAs would be available.

In summary, *the requirements imposed by the 1985 amendment apply only to Mark III and ice condenser containment plants and include the following:*

Analytical requirements

- *Type of accident, viz. degraded core accident with core remaining in-vessel,*
- *Source of hydrogen (fuel-cladding oxidation)*
- *Hydrogen source term (75% metal-water oxidation reaction)*

Physical requirements

- *Control system capable of mitigating hydrogen from 75% metal-water reaction*

3.3 Relationship of 10 CFR 50.44 to Other Regulations and Implementing Documents

10 CFR 50.44 either references or is referenced by other regulations. In addition, guidance is provided with the requirements in 10 CFR 50.44 or the associated regulations in the form of implementing documents such as regulatory guides, etc. These are summarized and described below.

3.3.1 Relationship of 10 CFR 50.44 to Other Regulations

Table 3-1 provides a list of the regulations referenced in 10 CFR 50.44 and those regulations that have some type of regulatory association with 10 CFR 50.44. The applicable or referenced section in 10 CFR 50.44 is also listed.

Table 3-1 Summary of Related Regulations

Applicable 10 CFR 50.44 Section	Referenced/ Related Regulation	Description of Requirement
(a)	10 CFR 50.82(a)(1)	Excludes from purview of 10 CFR 50.44 nuclear power reactor facilities that have certified permanent cessation of operation.
(b)(1)	10 CFR 50.47, App E	H2 monitors required by Emergency Response data system
	GDC 13	Instruments must be provided to monitor variables for accident conditions
	10 CFR 50.36	Tech Specs on monitor operability and testing
	GDC 43	Monitor testing
	10 CFR 21, App B	Procurement and QA for safety-grade monitors .
(b)(2)	GDC 41	Provide systems to control concentration of H2, O2 to insure containment integrity
	10 CFR 50.36	Tech Specs on mixing systems
(b)(3)	GDC 54,56	Requirements on containment penetrations for external recombiners and purge-repressurization systems
	App B	Quality standards for combustible gas control systems
	GDC 5	Sharing of external recombiners between units at one site
	10 CFR 50.36	Tech Spec requirements and surveillance testing of recombiners
	10 CFR 50.55a	ISI check valve tests
	App J	Testing of containment penetrations (App. J)
(c)(1)(ii)	GDC 50, 16	Containment shall accommodate, with sufficient margin, conditions resulting from a LOCA, including energy sources, as required by 10 CFR 50.44, from metal-water and other chemical reactions resulting from degradation but not total failure of ECCS. Containment shall establish leak-tight barrier against uncontrolled release to environment and assure conditions important to safety are not exceeded for duration of postulated accident.

Table 3-1 Summary of Related Regulations

Applicable 10 CFR 50.44 Section	Referenced/ Related Regulation	Description of Requirement
(c)(3)(ii)	GDC 54, 56	Applies to containment penetrations for external recombiners 54: provides requirements on piping systems penetrating containment 56: provides requirements on primary containment isolation
(c)(3)(iii)	Appendix A and B	Requirements for design of high point vents and associated controls, instruments, and power sources
(c)(3)(iv)	10 CFR 50.55a	ASME Codes for steel containments required to demonstrate structural integrity for Mark III and ice condenser plants
(c)(3)(vi)(A)	10 CFR 50.4	Specifies requirements for written communications from licensees operating Mark III and ice condenser plants that are required to submit accident analyses
(d)(1), (d)(2)	10 CFR 50.46(b)	Specifies maximum H2 generation in postulated LOCA for purposes of complying with ECCS acceptance criteria; referenced in original version of 10 CFR 50.44 as a basis for the design of the H2 control system for facilities in compliance with 10 CFR 50.46(b)
(f)	10 CFR 100.11(a)(2)	Pertains to facilities whose notice of hearing on CP application was published between 12/22/68 and 5/11/1970; applies dose based criteria, with doses calculated in accordance with the regulation in 10 CFR Part 100 used to develop exclusion area and LPZ boundary distances; if criteria are met, only purging system is necessary, if not, a second gas control system is required (repressurization system or a combined purge-repressurization system are acceptable);
	GDC 41,42,43	Both purge and repressurization systems have to comply with GDC 41, 42, 43; containment shall not be repressurized beyond 50% of design pressure
(g)	10 CFR 100.11(a)(2)	Pertains to facilities whose notice of hearing on CP application was published prior to 12/22/68; applies dose based criteria, with doses calculated in accordance with the regulation in 10 CFR 100 used to develop exclusion area and LPZ boundary distances; if criteria are met, only purging system is necessary, if not, a second gas control system is required (repressurization system or a combined purge-repressurization system are acceptable);

Table 3-1 Summary of Related Regulations

Applicable 10 CFR 50.4 4 Section	Referenced/ Related Regulation	Description of Requirement
	GDC 41,42,43	Both purge and repressurization systems have to comply with GDC 41, 42, 43; containment shall not be repressurized beyond 50% of design pressure
Note (1)	10 CFR 50.34 (f)	10 CFR 50.34 (f) “Additional TMI-related requirements” establishes requirements for combustible gas control for future plants whose applications for a construction permit or manufacturing license were pending as of 2/16/1982
Note (1): Regulations not explicitly referenced in 10 CFR 50.44, but are related to 10 CFR 50.44 requirements.		

10 CFR 50.82(a)(1), “Termination of License” (for power reactor licensees) requires that power reactor licensees who have decided to permanently cease operation must, within 30 days, submit a written certification to the NRC stating the date on which operations have ceased or will cease as required under 10 CFR 50.4(b)(8). Once fuel has been permanently removed from the reactor vessel, the licensee must submit a certificate to that effect to the NRC stating the date on which fuel was removed from the reactor vessel and the disposition of the fuel as required by 10 CFR 50.4(b)(9). Once these certifications have been submitted as required under 10 CFR 50.82(a)(1), then 10 CFR 50.44 ceases to apply to the reactor facility.

10 CFR 50.47 “Emergency Plans” and Appendix E “ Emergency Planning and Preparedness for Production and Utilization Facilities” also include the requirement stated in 10 CFR 50.44 (b)(1) to measure the hydrogen concentration in containment. Section VI “Emergency Response Data System” of Appendix E requires the licensee to provide accurate and timely updates of a limited set of parameters to the NRC Operations Center in the event of an emergency. Containment parameters to be supplied for PWRs include pressure, temperature, hydrogen concentration and sump levels. Containment parameters required to be provided for BWRs include drywell pressure, temperature and sump levels, hydrogen and oxygen concentrations, and suppression pool level and temperature.

General Design Criteria 13 of 10 CFR Part 50 Appendix A requires that instruments be provided to monitor variables for accident conditions as appropriate to assure adequate safety, including those variables that can affect the fission process, the integrity of the reactor core, the reactor coolant system boundary, and the containment and its associated systems.

10 CFR 50.36 requires establishment of a technical specification limiting condition of operation for installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. This requirement implies technical specifications on hydrogen monitor operability.

10 CFR Part 50 Appendix A "General Design Criteria for Nuclear Power Plants" and Appendix B "Quality Assurance Criteria for Nuclear Power Plants" apply to the design of the high point vents and the associated controls, instruments and power sources as required by 10 CFR 50.44 (c)(3)(iii).

10 CFR Part 50 Appendix A General Design Criteria 41 "Containment atmosphere cleanup" requires systems to control the concentration of hydrogen, oxygen, and other substances which may be released into the reactor containment following postulated accidents to assure that containment integrity is maintained. Hydrogen monitors fall within the purview of GDC 41.

GDC 42 "Inspection of containment atmosphere cleanup systems" contains requirements on the inspection of systems covered by GDC 41.

GDC 43 "Testing of containment atmosphere cleanup systems" imposes testing requirements on systems covered under GDC 41 which require periodic functional testing to assure operability of the systems as a whole.

10 CFR Part 21 imposes procurement requirements on safety-grade equipment. Since the hydrogen monitors are treated as safety-grade Class 1E electrical equipment they are subject to the requirements of Part 21.

10 CFR Part 50 Appendix B imposes quality assurance requirements on nuclear power plants systems and components.

The requirement (b)(2) in 10 CFR 50.44 is meant to insure a mixed atmosphere in containment. GDC 41, referred to above, requires systems to control the concentration of any releases into containment, including releases of hydrogen and oxygen, to assure containment integrity and thus applies to systems designed to provide a mixed atmosphere in containment. The functional types of systems provided vary by containment design. In some large dry containments, for example, this requirement is met by the containment spray system that promotes convective mixing of the containment atmosphere. Containment sprays are subject to technical specification requirements of 10 CFR 50.36.

Requirement (b)(3) in 10 CFR 50.44 calls for a capability to control combustible gas concentration in the containment following a postulated LOCA and the 1981 amendment to the original rule required via (c)(3)(ii) licensees that relied on purge/repressurization systems as the primary means for controlling combustible gas following a LOCA to install internal recombiners or a capability to install an external recombiner.

GDC 5 "Sharing of Structures, Systems, and Components" has provisions that apply to the sharing of external recombiners between different units at one site. Recombiners are subject to the technical specification and surveillance testing requirements of 10 CFR 50.36. In addition, ISI check valve tests have to be carried out on a quarterly basis.

10 CFR 50.55a "Codes and Standards" defines the inservice testing requirements for various plant systems and components and incorporates references to the requirements of the ASME Boiler and Pressure Vessel Code for steel containments that are required to be met in order to demonstrate containment structural integrity for Mark III and ice condenser containments under 10 CFR 50.44 (c)(3)(iv)(B)(1).

10 CFR Part 50 Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" contains requirements that apply to testing of containment penetrations and the quality standards of Appendix B apply to combustible gas control systems.

GDC 50 "Containment design basis" requires that the containment structure shall be designed to accommodate, with sufficient margin, pressure and temperature conditions resulting from a LOCA. The margin shall reflect consideration of energy sources, as required by 50.44, from metal-water and other chemical reactions resulting from degradation, but not complete failure, of emergency core cooling functioning.

GDC 16 "Containment design" requires that the reactor containment and associated systems shall establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that design conditions important to safety are not exceeded for the duration of the postulated accident.

10 CFR Part 50 Appendix A, General Design Criteria 54 "Piping systems penetrating containment" and General Design Criteria 56 "Primary containment isolation" apply to containment penetrations used for external recombiners (as well as containment penetrations for purge-repressurization systems) that may be installed by licensees as provided by 10 CFR 50.44 (c)(3)(ii)(A) and 10 CFR 50.44 (c)(3)(ii)(B). GDC 54 requires that piping systems penetrating primary reactor containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. GDC 56 requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with primary isolation valves.

10 CFR 50.4 "Written Communications" specifies requirements for all written communications from licensees operating Mark III and ice condenser containment plants that are required to submit analyses under the provisions of 10 CFR 50.44 (c)(3)(vi)(A). These analyses pertain to an evaluation of the consequences of hydrogen generated during an accident involving up to 75% of the clad metal-water reaction, include consideration of hydrogen control measures, include time period of recovery from the degraded condition, use accident scenarios supported by the NRC staff, and support the design of the selected hydrogen control system.

10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water power nuclear reactors" and Appendix K "ECCS Evaluation Models" establishes the amount of hydrogen generated in a postulated LOCA for the purposes of determining compliance with ECCS performance criteria. As mentioned above, the original version of the rule published as 10 CFR 50.44 based the design of the hydrogen control system on the 10 CFR 50.46 criteria for maximum hydrogen generation from the metal-water reaction with a factor of 5 added as a safety margin against unpredicted events during the evolution of the accident.

10 CFR Part 100 "Reactor Site Criteria" contains section 100.11 that provides a method for determining the distance of the exclusion area and low population zone boundary. 10 CFR 50.44 (f) refers to facilities whose notice of hearing of an application for a construction permit was published between 12/22/68 and 11/5/70. This requirement states that if the incremental dose from purging (and repressurization if a repressurization system is provided) occurring at all points beyond the exclusion area boundary after a postulated LOCA, calculated in accordance with 10 CFR 100.11(a)(2), is less than 2.5 rem whole body and less than 30 rem to the thyroid, and if the combined dose at the low population zone outer boundary from purging and the postulated LOCA calculated in accordance with 10 CFR 100.11(a)(2) is less than 25 rem whole body and 300 rem thyroid, then only a purging system is necessary. The purging system and any associated filtration systems are required to be designed in accordance with GDC 41, GDC 42 and GDC 43 . If the criteria are not met, then another combustible gas control is required which could be a repressurization system or a combined purge-repressurization system. The second system also has to comply with the requirements of GDC 41, 42, and 43.

10 CFR 50.44 (g) applies similar requirements to facilities whose notice of hearing of an application for a construction permit was published on or before 12/22/68.

10 CFR 50.34 "Contents of applications; technical information" that deals with applications for a construction permit has a section, 10 CFR 50.34 (f) "Additional TMI-related requirements", that applies to combustible gas control. Part 50.34 (f) applies to applicants for a LWR construction permit or manufacturing license whose application was pending as of February 16, 1982.

Several paragraphs of this section pertain to hydrogen control measures. Paragraph (f) (2) (ix) requires applicants to provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Applicants are asked to perform an evaluation of alternative hydrogen control systems that would meet this criterion including, as a minimum, hydrogen ignition systems and a post-accident inerting system. The evaluation should include: (a) a comparison of the costs and benefits of the alternatives considered, (b) for the selected system, analyses and test to verify compliance with the performance required, (c) for the selected system, preliminary design descriptions of equipment, function, and layout.

Only preliminary design information on the tentatively preferred option among the alternatives considered is required at the construction permit stage. However, the regulation requires that the hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(a) Hydrogen concentrations uniformly distributed in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion,

(b) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could lead to loss of containment integrity or loss of appropriate mitigating features,

(c) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety functions during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system,

(d) If the method selected for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

This regulation clearly goes well beyond the design basis LOCA hydrogen generation specified in 10 CFR 50.46 for compliance with ECCS acceptance criteria and modified with a safety margin factor of 5 in 10 CFR 50.44 to serve as a basis for design of hydrogen control systems. In 10 CFR 50.44, plants with Mark III and ice condenser containments are required to demonstrate hydrogen control systems that can mitigate the amount of hydrogen generated by the equivalent of a 75% clad metal-water reaction. However, 10 CFR 50.34 (f) requires plants submitting applications in the post-1982 period to have the capability of handling hydrogen generated from the equivalent of a 100% clad metal-water reaction and also to ensure that hydrogen concentrations in containment do not exceed 10% during and following an accident.

3.3.2 Relationship of 10 CFR 50.44 to Implementing Documents

Guidance that is provided to the licensee in meeting the requirements of 10 CFR 50.44 via the various implementing documents are summarized in Table 3-2 and described below. The applicable section of 10 CFR 50.44 and related regulation is also listed.

Table 3-2 Summary of Implementing Documents

Applicable 50.44 Section	Referenced Document	Description of Guidance
(a)(1)(a)(2) (a)(3)(d)(1) (d)(2)	RG 1.7	Provides guidance on H2 generation following a postulated LOCA, from post-accident radiolysis of water, and metal corrosion.
(b)(1)	RGs 1.70,1.89,1.97 SRP Sec 6.2.5, NUREGs-0737, 0718, 0660 RG 1.97, ANSI-ANS-4.5 RG 1.118	Provide guidance on design bases, system designs, and design evaluation of systems to monitor combustible gas concentrations within containment regions Guidance on instrumentation to assess plant conditions during an accident, establishes H2 concentration in containment or drywell as a Type C variable, recommends H2 monitors as safety-grade Guidance on monitor testing requirements (RG 1.118)
(b)(2)	RG 1.70, SRP 6.2.5	Guidance on design bases, system design, and evaluation of mixing systems
(b)(3) (c)(3)(ii) (d)(1) (d)(2)	RG 1.7 RG 1.70, SRP 6.2.5 NUREG-0737, NUREG-0578, GL 83-02, SECY 80-399 ASME Section XI RG 1.26, SRP 6.2.5 RG 1.29, SRP 6.2.5 GL 84-09 NUREG-0737 RG 1.52, GL 83-13	Guidance on H2 generated in metal-water reaction, radiolysis, corrosion; Design and evaluation of systems to reduce combustible gas concentrations Dedicated penetrations for external recombiners or purge systems Penetration piping leakage surveillance Quality standards for design, fabrication, erection, and testing Designed for SSE For inerted Mark I containments with NOHC<11/5/70 that do not rely on purge-repress systems as primary means of H2 control, recombiners not required provided certain TS are met Containment atmosphere dilution systems considered to be purge systems Design, testing and maintenance criteria for post-accident ESF atmosphere cleanup systems

Table 3-2 Summary of Implementing Documents

Applicable 50.44 Section	Referenced Document	Description of Guidance
(c)(3)(iii)	RG 1.92, RG 1.100, IEEE 344-1975 NUREGs-0737,0660	Seismic qualification and EQ of vent systems Guidance on vent system design
(c)(3)(i)	ASME Section XI	Inerting system lines penetration piping leakage surveillance requirements
(c)(3)(iv)(A)	ASME B&PV Code sections	ASME B&PV code sections for steel containment (Section III, Subsubarticle NE-3220, Service Level C limits); ASME B&PV Code sections for concrete containments (Section III, Subsubarticle CC-3720, Factored Load Category)

Regulatory Guide 1.7 [7] "Control of Combustible Gas Concentrations in Containment Following a LOCA" Rev. 2, Nov. 1978 provides guidance on the implementation of the original version of 10 CFR 50.44 in LWRs with zircaloy clad fuel. As can be surmised from the date of this Regulatory Guide, the implementation guidance deals only with the part of 50.44 which made up the original rule, i.e., hydrogen generation as a result of a LOCA.

This guide references GDC 35 (emergency core cooling), GDC 50 (containment design basis) and GDC 41 (containment atmosphere cleanup). It refers to the "new" 50.44 and states that the guide provides methods for implementing the new regulation.

After a LOCA, hydrogen can result from: (1) metal-water reaction in which the zirconium clad oxidizes and hydrogen is evolved by Zr + 2H20 = ZrO2 + 2H2, (2) post-accident radiolysis of water by released fission products in solution which will lead to both H2 and O2 being evolved, and (3) corrosion of metals inside containment.

If enough hydrogen is generated it can react with O2 in the containment. If the H2-O2 reaction is rapid it can cause high temperatures or pressures and either breach containment or cause leakage above the technical specifications and also potentially damage safety SSCs.

The extent of metal-water reaction and the amount and rate of hydrogen produced depends on the assumptions underlying accident evolution and the effectiveness of emergency core cooling systems (ECCS). The guide references ECCS analytical models described in 36 FR 12248 of June 29, 1971 and amended in 36 FR 26042 of Dec. 18, 1971; and in the record of the rulemaking hearing, Docket RM-50-1, which led to the issuance of Part 50.46 acceptance criteria for ECCS.

The maximum amount of metal-water reaction, following a postulated LOCA, allowed by ECCS acceptance criteria in Part 50.46 is 1% of the cladding mass.

To establish the design basis for containment gas control systems, the guide recommends that the amount of hydrogen should be based on that calculated for establishing compliance with Part 50.46 but should also include a safety margin. This margin is set as at least 5 times the amount calculated for compliance with Part 50.46. However, the guide concedes that this calculated amount could be small for many plants as a result of other requirements for ECCS contained in

50.46. So a lower limit for the amount of hydrogen generated following a postulated LOCA is also recommended in the guide. This is based on the criterion of the 1% of cladding mass reacting in the metal-water reaction. However, in order not to penalize fuel with thicker cladding (since cladding oxidation is a surface phenomenon), a criterion based on hydrogen generated per unit cladding area was selected by specifying a hypothetical uniform depth of cladding surface reacted. This hypothetical depth was based on 1% of the thickness of the thinnest fuel cladding (0.023") used at the time the guide was issued.

Thus, to comply with 10 CFR 50.44, the hydrogen generated after a LOCA should be 5 times the maximum amount calculated for purposes of compliance with Part 50.46 but not less than the amount generated from a reaction of cladding metal to a depth of 0.00023". (Safety Guide 7, the precursor to RG 1.7, recommended that hydrogen control systems be designed for a 5% metal-water reaction.)

The rate of hydrogen production in the metal-water reaction is assumed to occur on the following basis: the initial reaction takes place over a short period of time early in the LOCA, near the end of the blowdown and the core refill phases of the transient. Since the duration of the blowdown and refill phases is of the order of several minutes, it is assumed that hydrogen will be generated over a 2-minute period, which represents the period of time during which the maximum heatup occurs, at a constant rate. Further, the hydrogen will mix with the steam released from the RCS and be distributed uniformly over the containment volume.

RG 1.70 [8],"Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Section 6.2.5, "Combustible Gas Control in Containment" provides guidance on the design bases, system design, and design evaluation of systems to mix the containment atmosphere, monitor combustible gas concentrations within containment regions, and reduce combustible gas concentrations in containment. RG 1.70 references GDC 41 that requires provisions of systems to control concentrations of hydrogen and oxygen released into containment from postulated accidents.

NUREG-0800 [9] "Standard Review Plan" Section 6.2.5 provides guidance to the NRC staff on reviewing the portion of the SAR dealing with the production and accumulation of combustible gases in containment following a design basis LOCA, the capability to monitor combustible gas concentrations in containment, the capability to mix the combustible gas concentrations within the containment atmosphere, and the capability to reduce the combustible gas concentrations in containment by suitable means such as purging, dilution or recombination.

Regulatory Guide 1.97 [10] "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" establishes that hydrogen concentration in the containment and drywell is a Type C variable (i.e., a variable that provides information to the control room operator to indicate the potential for breach or actual breach of the barriers to fission product release). Monitoring of hydrogen concentration is needed in BWRs to detect the potential for breach and accomplishment of mitigation, and in PWRs for detection of the potential for breach, accomplishment of mitigation, and long-term surveillance. Hydrogen monitors in containment are classified safety-grade (Class 1E) based on RG 1.97 recommendations. In the post-TMI period, an ANSI/ANS-4.5 Standard was proposed classifying hydrogen concentration in containment as Type C variable.

Regulatory Guide 1.118 [11] "Periodic Testing of Electric Power and Protection Systems" provides guidance on the periodic testing of electric power and protection systems that are classified as safety-grade systems.

Regulatory Guide 1.89 [12] "Environmental Qualification of Certain Electronic Equipment Important to Safety for Nuclear Power Plants" contains guidance on the environmental qualification of electrical equipment important to safety.

Regulatory Guide 1.52 [13] "Design, Testing and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" provides guidance on design, testing, and maintenance criteria for post-accident engineered safety features (ESF) of containment atmosphere cleanup systems including HEPA air filters and charcoal adsorption units.

Generic Letter GL 83-13 [14] provides clarification of surveillance requirements for HEPA filters and charcoal adsorber units in standard technical specifications on ESF cleanup systems. In the letter documenting the results of the NEI survey [3], RG 1.52 and GL 83-13 are mentioned by licensees operating plants with large dry PWR containments and BWR Mark I containments as the guidance documents they use in complying with 50.44 (b)(3).

Regulatory Guide 1.92 [15] "Combining Modal Responses and Spatial Components in Seismic Response Analysis" and Regulatory Guide 1.100 [16] "Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants" contain guidance on seismic qualification that is incorporated by reference in section II.B.1 of NUREG-0737 that provides the requirements for the reactor coolant system high-point vent designs. Environmental qualification of the vents are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

NUREG-0737 [17] "Clarification of TMI Action Plan Requirements" contains several sections relevant to 50.44 requirements. These include section II.B.1 on reactor coolant system vents, section II.B.3 on post-accident sampling capability that calls for sampling of hydrogen levels in the containment atmosphere, section II.E.4.1 on dedicated hydrogen penetrations concerning containment penetration systems for external recombiners or purge systems, and section II.F.1 on containment hydrogen monitors.

NUREG-0737 incorporates, by reference, NUREG-0578 [18] "TMI-2 Lessons Learned Task Force Report and Short-Term Recommendations" of July 1979, NUREG-0660 [19] "NRC Action Plan Developed as a Result of the TMI-2 Accident", August 1980, Regulatory Guide 1.26 [20] on quality standards for design, fabrication, erection, and testing, Regulatory Guide 1.29 [21] on seismic classification, and IEEE 344-1975 on environmental qualification.

Generic Letter 84-09 [6] addresses the recombiner capability requirements of 50.44 (c)(3)(ii) and is directed at BWR Mark I plants for which notices of hearing on applications for construction permits were published before November 5, 1970. For these plants, given the inerting requirements under 50.44(c)(3)(i), it was determined that purging/repressurization systems were not the primary means for controlling combustible gas concentrations following a LOCA. Hence, these plants were exempted from providing recombiners as required under 50.44(c)(3)(ii) subject to meeting certain criteria related to technical specifications on controlling oxygen concentrations in containment. [Under 50.44 (e), plants whose notices of hearing on applications for a construction permit were published on or after November 5, 1970 were not permitted to use purging and/or repressurization systems as the primary means of controlling combustible gases following a postulated LOCA but instead had to install means such as recombiners that would not lead to a significant release from containment].

3.4 Current Industry Implementation of 50.44

Six high-level requirements imposed by 50.44 have been identified above: (1) Measuring hydrogen in containment (established hydrogen monitors) [(b)(1)], (2) Systems to insure mixed containment atmosphere [(b)(2)], (3) Systems to control combustible gases [(b)(3) and (c)(3)(ii)], (4) High-point vents on the reactor coolant system [(c)(3)(iii)], (5) Inerting of Mark I and II containments [(c)(3)(i)], and (6) Installation of a hydrogen control system to deal with a 75% metal-water reaction in Mark III and ice condenser containments [(c)(3)(iv)..(vii)].

In order to understand the basis on which the industry is implementing the high-level requirements of 50.44, an effort is underway to obtain implementation data from the industry. So far, preliminary information has been obtained from two sources: (1) a review of NRC documents for three specific plants, Grand Gulf 1, Nine Mile Point 2, and Sequoyah 1 and 2, and (2) an NEI survey of licensees to determine the sources of guidance, regulatory, industry, or specific utility, being used to assure implementation of the requirements. Responses to this survey were received from 23 units (PWR large, dry containments) and 10 units (BWR Mark I containments).

Table 3-3 below summarizes this preliminary information received on the systems used to implement the requirements and the special treatment of these systems. Since the sources of information are limited, the data in the table may not reflect implementation practices across the nuclear industry. (No information was received regarding high point vents.)

Table 3-3 Basis of Industry Implementation of 50.44 High-Level Requirements

50.44 requirement	Industry implementation based on review of 3 plants	Guidance/Implementing documents identified in NEI survey
(b)(1) Measure H2 in containment	H2 monitor identified as essential equipment needing safety grade treatment in all 3 plants surveyed. (NUREG-0831)	RG 1.7, RG 1.97, NUREG-0737
(b)(2) Mixed containment atmosphere	***PWR ice condenser: Sequoyah*** For DBA conditions, mixing requirements are met by the air return fans, which are safety grade engineered safety features. For degraded core accidents, EPRI tests will fans operable showed good mixing results. Staff concluded that formation of significant hydrogen concentration gradients in containment is unlikely under these conditions and that detonable pockets will not occur given operation of the mixing and igniter systems operating at the lower hydrogen flammability limit.	NUREG-0800, Reg Guide 1.70, Tech Specs
	BWR Mark III: Grand Gulf For DBA mixing requirements there are no active fan system; codes predict adequate mixing based on differential pressures. For degraded core accidents mixing was confirmed by licensee analysis. Bounding detonation calculation on volume below HCU floor also showed that containment could withstand the loading. (NUREG-0831)	

Table 3-3 Basis of Industry Implementation of 50.44 High-Level Requirements

50.44 requirement	Industry implementation based on review of 3 plants	Guidance/Implementing documents identified in NEI survey
	BWR Mark II: NMP2 Before initiation of the recombiner, the drywell and suppression chamber will be mixed as a result of natural convection arising from temperature differences between the atmosphere and primary containment walls and molecular diffusion. Mixing is further promoted by blowdown of steam and water out the broken pipe and operation post-accident of containment sprays. The combustible gas control system also mixes the primary containment atmosphere. (SRP 6.2.5, NUREG-1047)	
(b)(3) and (c)(3)(ii) Combustible gas control systems, e.g. recombiners	Recombiners are identified as safety grade in the plants surveyed on the basis of RG 1.7. ***BWR Mark II: NMP2*****:** Recombiner system is 100% redundant, essential equipment, seismic Category I, safety class 2. In addition, a back-up containment purge capability is provided in accordance with Reg. Guide 1.7 which is used in conjunction with the standby gas treatment system. The applicant calculated that the set point for recombiner operation is not reached for 2.75 days. (SER 2/83, NUREG-1047)	RG 1.7, RG 1.52, ANSI N510, GL 83-13, NUREG-0737, NUREG-0800
(c)(3)(I) Inert Mark I and II	***BWR Mark II: NMP2*****:** Plant has a non-seismic Category I liquified nitrogen storage and gas distribution system which limits the oxygen concentration to 4% volume when inerted. (NUREG-1047, SER Feb. 1983)	NA
(c)(3)(iv)(v)(vi)(vii) H2 control system for 75% m-w reaction	***PWR ice condenser: Sequoyah:*** Permanent Hydrogen Mitigation System (PHMS) consists of 64 igniters distributed throughout the upper, lower, and ice condenser compartments. Testing programs in conjunction with Duke Power and American Electric Power were conducted to demonstrate the ability of the system to mitigate the hydrogen threat. CLASIX calculations also performed. Staff required 4 additional igniters as part of SER. Confirmatory analyses performed using CSQ code. Modified COMPARE code used to evaluate containment response. (NUREG-0011, Supplement 6, 12/82)	NUREG-0737

Table 3-3 Basis of Industry Implementation of 50.44 High-Level Requirements

50.44 requirement	Industry implementation based on review of 3 plants	Guidance/Implementing documents identified in NEI survey
	BWR Mark III: Grand Gulf: Hydrogen Ignition System (HIS) installed consisting of 90 igniters distributed throughout the drywell, wetwell, and upper compartment. EPRI sponsored test program for BWR-6/Mark III owners including operability testing and combustion testing. Calculations were performed with CLASIX-3/MARCH codes. (NUREG/CR-2530)	

3.5 Implementation of 50.44

The current implementation of 10 CFR 50.44 is summarized in Tables 3-4 through 3-9 below. This table summarizes the information from Sections 3.2.and 3.4 above. While the sources listed in the tables may not be all exhaustive, they trace the implementation of each specific requirement in the regulation down to practical level of detail.

In each table, for each high level requirement, the supporting requirements contained within 10 CFR 50.44 itself are provided. Additional regulatory requirements, which are not contained in 10 CFR 50.44 itself, but which support the regulation are next provided. Finally, the guidance contained in implementing documents such as Regulation Guides, NUREGs, Sections of the Standard Review Plan, industry codes and standards and other supporting documents, are provided.

3.5.1 Measuring Hydrogen Concentration in Containment

The majority of plants have implemented the requirement for measuring hydrogen concentration in containment by installing continuous safety-grade monitors. These monitors are also credited with meeting the emergency response requirements of 50.47(b)(9) and Part 50 Appendix E. The basis for this implementation is: (1) the recommendation in Regulatory Guide 1.7 that systems to measure combustible gases in containment should meet the requirements for an engineered safety feature, and (2) a post-TMI requirement stated in NUREG-0737, Item II.F.1 that requires all plants to provide a continuous indication of hydrogen concentration in the containment for accident monitoring. This requirement imposed the design and quality criteria of Regulatory Guide 1.97 that treats hydrogen monitors as redundant, safety-grade, Class 1 E electrical equipment and also required that monitors be included in the plant technical specifications. This information is summarized below in Table 3-4.

Table 3-4 Summary of Requirements and Guidance for Measuring Hydrogen Concentration in Containment

Supporting Requirements
NONE
Related Regulatory Requirements
• H2 monitors (50.47, Part 50 App E) • Instruments to monitor variables for accident conditions (GDC 13 Part 50 App A) • Technical Specifications on monitor operability and surveillance testing (50.36) • Monitor testing reqmts (GDC 43) • Monitor (safety-grade) procurement and QA reqmts (10 CFR 21, App B)
Supporting Guidance
• Guidance on H2 monitoring, system design bases, evaluation, and classification (RG 1.70, RG 1.97, RG 1.89, SRP 6.2.5, NUREG-0737, NUREG-0718, NUREG-0660, ANSI-ANS 4.5) • Guidance on testing requirements (RG 1.118)

3.5.2 Ensuring a Mixed Containment Atmosphere

In most plants, systems that ensure mixing of the containment atmosphere are the same as those providing containment heat removal. In plants that have active systems for accomplishing the mixing function, such as air return fans or sprays, licensees have predominantly implemented this requirement by treating these systems as engineered safeguard features on the basis of the design criteria of GDC 41, the guidance contained in Regulatory Guide 1.70, and the provisions of the Standard Review Plan (NUREG-0800). These systems are included in the plant technical specifications. This information is summarized below in Table 3-5.

Table 3-5 Summary of Requirements and Guidance for Ensuring a Mixed Containment Atmosphere

Supporting Requirements
NONE
Related Regulatory Requirements
• Systems to control conc. of H2 & O2 to insure containment integrity (GDC 41) • Tech Specs on mixing systems (50.36)
Supporting Guidance
• Guidance on design bases and evaluation of mixing systems (RG 1.70, SRP 6.2.5)

3.5.3 Control of Post LOCA Combustible Gases

A majority of licensees have complied with the requirement to provide control of post-LOCA combustible gases by installing safety-grade internal recombiners. The recombiners are treated as an engineered safeguard feature and essential equipment on the basis of the guidance provided in Regulatory Guide 1.7 and included in the plant technical specifications. Older plants, whose notice of hearing on a construction permit was received prior to 11/5/1970, are allowed to have only purge/repressurization systems as a primary means of combustible gas control. However these systems require dedicated containment penetrations per NUREG- 0737 that are subject to the testing requirements of Appendix J. The latter requirements also apply to containment penetrations in plants that use external recombiners. This information is summarized below in Table 3-6.

Table 3-6 Summary of Requirements and Guidance for Control of Post LOCA Combustible Gases

Supporting Requirements
• Following LOCA show: no uncontrolled H2-O2 recombination or plant could withstand consequences; if not, inert containment (c)(1)(i),(c)(1)(ii),(c)(2) • If purge/repress. systems are *primary* means of control, provide recombiners; assume H2 equal to 5% metal-water reaction or 5x that needed to comply with 50.46 (c)(3)(ii),(d)(1),(d)(2) • Containment penetrations for ext. recombiners and purge/repressurization systems (c)(3)(ii)(A),(c)(3)(ii)(B) • If NOHC received — > 11/5/70 require systems other than purge-repress. as primary means of comb gas control (e) — < 11/5/70 require only purging systems if certain dose based requirements calculated on basis of 100.11 are met (f,g)

Table 3-6 Summary of Requirements and Guidance for Control of Post LOCA Combustible Gases

Related Regulatory Requirements
• Amount and rate of H2 generated in LOCA (50.46) • Reqmts. on containment penetrations for ext. recombiners and purge-repress. systems (GDC 54, 56) • Quality standards for comb gas control systems (App B) • Dose calculation methods for 50.44(f,g) compliance (100.11) • Sharing of external recombiners between units at one site (GDC 5) • Tech Spec requirements and surveillance testing of recombiners (50.36) • ISI check valve tests (50.55a) • Testing of containment penetrations (App J)
Supporting Guidance
• Guidance on H2 generated in metal-water reaction, radiolysis, corrosion (RG 1.7) • Design and evaluation of systems to reduce comb gas concentrations (RG 1.70, SRP 6.2.5) • Dedicated penetrations for ext. recombiners or purge systems (NUREG-0737, NUREG-0578, GL 83-02, SECY 80-399) • Penetration piping leakage surveillance (ASME section XI) • Quality standards for design, fabrication, erection, and testing (RG 1.26, SRP 6.2.5) • Designed for SSE (RG 1.29, SRP 6.2.5) • For inerted Mark I containments with NOHC<11/5/70 that do not rely on purge-repress systems as primary means of H2 control, recombiners not required provided certain TS are met (GL 84-09) • Containment atmosphere dilution systems considered to be purge systems (NUREG-0737) • Surveillance reqmts for HEPA filters and charcoal adsorbers in TS on ESF cleanup systems (RG 1.52, GL 83-13)

3.5.4 RCS High Point Vents

All licensees have implemented this requirement on the basis of the post-TMI requirements identified in NUREG-0737, section II.B.1, that specify the quality assurance and design criteria for the vents on the reactor coolant system. This information is summarized below in Table 3-7.

Table 3-7 Summary of Requirements and Guidance for RCS High Point Vents

Supporting Requirements
• vents for the RCS, reactor vessel head and for other systems • remotely operated from control room • conform to Appendix A and B • ensure low probability of failure and inadvertent or irreversible actuation • not aggravate the challenge to the containment or the course of the accident

Table 3-7 Summary of Requirements and Guidance for RCS High Point Vents

Related Regulatory Requirements
• Requirements for design of vents and associated systems (App A, App B) • Vent size smaller than LOCA definition (App A)
Supporting Guidance
• Seismic qualification and EQ of vent systems (IEEE 344-1975, RG 1.100, RG 1.92, CLI-80-21) • Guidance on vent system (NUREG-0737, NUREG-0660)

3.5.5 lnerting Mark I and Mark 11 Containments

Mark I and Mark II containment plants have met this requirement by installing nitrogen inerting systems whose containment penetrations meet the requirements of GDC 54 and 56 and are periodically tested as per provisions of Appendix J. This information is summarized below in Table 3-8.

Table 3-8 Summary of Requirements and Guidance for Inerting Mark I and II Containments

Supporting Requirements
NONE
Related Regulatory Requirements
• Inerting system lines that penetrate containment must meet redundancy and single-failure criteria (GDC 54, 56) • Testing of containment penetrations (App J) • Tech specs on inerting systems (50.36)
Supporting Guidance
• Penetration piping leakage surveillance (ASME section XI)

3.5.6 Requirements for Hydrogen Control System for 75% Metal-water Reaction

Licensees operating Mark III and Ice Condenser containment plants have met this requirement by installing a control system consisting of distributed hydrogen igniters that are powered by at least two separate and independent AC power sources. Licensees have also utilized analytical codes acceptable to NRC staff to demonstrate that the installed system can mitigate the amount of hydrogen generated in a 75% metal-water reaction. This information is summarized below in Table 3-9.

Table 3-9 Summary of Requirements and Guidance for Hydrogen Control System

Supporting Requirements
• Demonstrate containment structural integrity based on actual material properties or ASME B&PV code (c)(3)(iv)(B) • For H2 control system using post-accident inerting show containment can withstand increased pressure during the accident or following inadvertent full inerting in normal operation (c)(3)(iv)(D) • Reqmts. on systems and components for plants with post-accident inerting control systems (c)(3)(iv)(E) • Reqmts. on systems and components for plants that do not rely on inerting for H2 control (c)(3)(v)(A) • For plants with CP issued <3/28/79 provide evaluation of consequences of H2 using accident scenarios acceptable to NRC that support design of control system (c)(3)(vi)(A), (c)(3)(vi)(B)
Related Regulatory Requirements
• Reference to ASME B&PV code reqmts. for steel containments (50.55) • Written communications on accident analyses (50.4)
Supporting Guidance
• ASME B&PV code sections for steel containment (Section III, Subsubarticle NE-3220, Service Level C limits) • ASME B&PV Code sections for concrete containments (Section III, Subsubarticle CC-3720, Factored Load Category)

NOHC = Notice of hearing on application for construction permit
CP = construction permit
SSE = Safe Shutdown Earthquake

3.6 References

1. USNRC, Secy-99-264 “Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50,” November 8, 1999.

2. USNRC, “Options for Risk-Informed Revisions to 10 CFR PART 50 - “Domestic Licensing of Production and Utilization Facilities,” SECY-98-300, December 23, 1998.

3. Drouin, M. (NRC), memorandum to M. Cunningham (NRC), Subject: Transmittal of Results of Informal Nuclear Energy Institute (NEI) Survey on Benefits from Improving 10 CFR 50.44.”

4. Letter from Southern California Edison to NRC, Docket Nos. 50-361 and 50-362, September 10, 1998.

5. “Summary of Information Presented at an NRC-Sponsored Public Workshop on Options for Risk-Informed Revision to 10 CFR Part 50, September 15, 1999, Rockville, Maryland, “Sandia National laboratories,” January 2000.

6. USNRC, Generic Letter 84-09, “Recombiner Capability Requirements of 10 CFR 50.44 (c) (3) (ii),” May 8, 1984.

7. USNRC, “Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident,” Regulatory Guide 1.7, Revision 2, November 1978.

8. USNRC, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),“ Regulatory Guide 1.70, Revision 3, November 1978.

9. USNRC “Standard Review Plan for the Review of Safety Analysis Report for Nuclear Power Plants (LWR Edition),” NUREG-0800, Revision 2, July 1981.

10. 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.

11. USNRC “Periodic Testing of Electric Power and Protection Systems,” Regulatory Guide 1.118, Revision 3, April 1995.

12. USNRC, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Regulatory Guide 1.89, Revision 1, June 1984.

13. USNRC, “Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, “ Regulatory Guide 1.52, Revision 2, March 1978.

14. USNRC, Generic Letter 83-13, “Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications on ESF Cleanup Systems,” March 2, 1983.

15. USNRC, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,“ Regulatory Guide 1.92, Revision 1, February 1976.

16. USNRC, “Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants,” Regulatory Guide 1.100, Revision 2, June 1988.

17. USNRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737, November 1980.

18. USNRC, “TMI-2 Lessons Learned Task Force Report and Short-Term Recommendations,” NUREG-0578, July 1979.

19. USNRC “NRC Action Plan Developed as a Results of the TMI-2 Accident,” NUREG-0660, August 1980.

20. USNRC, “Quality Group Classifications and Standards for Water-, Steam-, and Radiative-Waste-Containing components of Nuclear Power Plants,” Regulatory Guide 1.26, Revision 3, February 1976.

21. USNRC, “Seismic Design Classification,” Regulatory Guide 1.29, Revision 3, September 1978.

4. RISK SIGNIFICANCE OF COMBUSTIBLE GASES

4.1 Concern Related to Combustible Gases

Combustible gas (namely hydrogen (H2)) can be generated, to a varying extent, in light water reactor (LWRs) during normal plant operation, design basis accidents (DBAs) and accidents involving extensive damage of the reactor core. In addition, during an accident involving extensive core damage, if the core melts through the reactor vessel and interacts with concrete CO can also be formed. Any accident initiator (i.e., loss of coolant, transient, loss of offsite power etc.) coupled with additional system or component failures can result in loss of coolant inventory, and thereby, lead to extensive core damage and hence the generation of large quantities of combustible gases. This concern therefore potentially applies to all core melt accident sequences.

During normal plant operation combustible hydrogen gas can be generated by radiolytic decomposition of the reactor coolant (i.e., water). However because of the configuration (i.e., closed cycle) of the reactor coolant system, hydrogen generation reaches an equilibrium condition and is not released to the containment atmosphere. Hydrogen generation is therefore not a concern in terms of containment integrity during normal plant operation.

In design basis loss of coolant accidents (LOCAs) the reactor core is predicted to be without coolant flow for a relatively short period of time. During this time the reactor core can reach temperatures high enough for the zircaloy cladding to oxidize in a steam environment. This oxidation is exothermic and produces hydrogen gas as a reaction product. In addition, an assumed release of radionuclides to containment, produces hydrogen via radiolytic decomposition of water. However the amount of hydrogen produced by these processes is relatively small and thus is not of major concern in terms of maintaining containment integrity. Hydrogen generation during design basis LOCAs can therefore be accommodated by relatively low capacity systems (such as recombiners and/or purge systems).

Accidents involving extensive core damage can be classified as degraded core or full core melt accidents. A degraded core accident involves extensive core damage (and melting of some constituents of the core) but the emergency core cooling system (ECCS) is restored in sufficient time to reflood the core and terminate the accident progression with the core retained in the reactor vessel. Using this terminology the accident at TMI-2 would be termed a degraded core accident. This definition, however, should not be confused with some earlier definitions of a degraded core accident that were limited to events in which the core was damaged but melting did not occur.

The main source of hydrogen generation for degraded core accidents is clad oxidation. The TMI-2 accident resulted in significant core damage and extensive clad oxidation (approximately 45% of the cladding) which generated a large quantity (400kg) of hydrogen. A significant quantity of hydrogen was released to containment and did ignite and burn (i.e. a deflagration). The resulting pressure pulse however was below the containment design pressure and did not challenge containment integrity.

The TMI-2 containment is a "large volume" design, which relies on a large free volume and a relatively high design pressure to mitigate the steam released during a design basis LOCA. Containments of this design therefore have a significant capacity for withstanding the pressure loads associated with combustion. This is also true for plants with subatmospheric containments, which have large internal volumes and high design pressures. However other containment designs (PWR ice condenser and BWR Mark I, II and III containments) rely on pressure suppression concepts (i.e., ice chests or water pools) to condense the steam released during a design basis LOCA. Pressure suppression containments therefore have smaller containment volumes and in some cases lower design pressures than large volume or subatmospheric containments. Consequently the smaller volumes and lower design pressures associated with pressure suppression containment designs makes them more vulnerable to hydrogen 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.

In a full core melt accident, the ECCS is not restored in time to prevent the damaged core from relocating into the bottom of the reactor vessel and melting through the lower vessel head. At this time several interactions can occur depending on the pressure in the reactor vessel and on conditions in the reactor cavity (i.e., flooded or dry). If the vessel is at high pressure the high temperature core debris can be dispersed as particles into the containment atmosphere. This is called high pressure melt ejection (HPME). During HPME particles can then directly heat the containment atmosphere, generate more hydrogen, and ignite any hydrogen in the containment atmosphere. This phenomena is termed direct containment heating (DCH). If the cavity is flooded the high temperature core debris could contact water in the reactor cavity. Under these circumstances the resulting fuel-coolant interactions (FCI) can generate significant quantities of steam and hydrogen very rapidly, which should be considered when formulating a realistic combustible gas source term. If the vessel is at low pressure and the cavity is dry the high temperature core debris can will interact with concrete in the region below the reactor vessel. Core-concrete interactions (CCI) can generate additional hydrogen from metal-water reactions (cladding and steel) and other non condensible (carbon dioxide) and combustible (carbon dioxide) gases. Limestone concrete generates significant quantities of steam, H2, CO2 and CO during CCI, whereas basalt concrete generates mostly steam and H2.

Containment failure during an accident involving a severely damaged core can lead to the release of a large quantity of radionuclides. The magnitude of the release depends on several containment related factors:

1. the size of the break in containment
2. whether or not the sprays are operating (enhanced aerosol deposition)
3. whether or not the release path passes through a pool of water (aerosol scrubbing), and
4. the time of release relative to time the radionuclides are released from the damaged reactor fuel

Therefore, not all containment failures lead to large releases. In order for the release to be large, the break has to be greater than 100 times the design basis leakage, the sprays should not be operating, and the release path should not be flooded. In addition, containment failures close to the time of reactor vessel melt-through have the potential to release more radionuclides (less time for natural aerosol deposition) than containment failures several hours after the onset of core damage. Failures close to the onset of core damage (less time for evacuation of the population) are, therefore, usually more important contributors to acute health effects. These failure modes are therefore important contributors to the large early release frequency (LERF) which is defined in Chapter 2. However, some late failures (within approximately 24 hours of the onset of core damage) can release large quantities of radionuclides. A definition of a large late release is also provided in Chapter 2.

The intent is to reduce the likelihood of generating significant quantities of combustible gases and to prevent (or control) the combustion of these gases if they should be generated during an accident involving severe damage to the reactor core.

4.2 Risk From Combustible Gases

In this section the evolution of knowledge regarding the generation and behavior of combustible gases is discussed. This evolution is summarized in Table 4-1 in terms of events and research activities that have influenced the regulations.

Table 4-1 Evolution of Knowledge Regarding Risk From Combustible Gases

TIME	EVENT	REGULATORY RESPONSE
1960s/ 1970s	• Core melt accidents not considered credible	• no regulations imposed but designers, operators and regulators of LWRs recognized the potential for generating H2 and regulatory guidance was provided
1975	• Reactor Safety Study (WASH-1400) • Accidents (e.g., transients) other than loss of coolant accidents contribute to the total core damage frequency (CDF) • Large quantities of combustible hydrogen (H2) gas predicted due to cladding oxidation but containment failure dominated by failure modes other than combustion • Large quantities of carbon dioxide (CO2) gas predicted from core-concrete interactions which contribute to containment failure by overpressurization • Reduction of carbon dioxide to combustible carbon monoxide (CO) not modeled	• no regulations imposed a consequence of WASH-1400
late 1970s	• Hydrogen generation predicted from design basis loss of coolant accidents (LOCAs)	• original version of 50.44 — measure concentration — mixed atmosphere — control concentration
1979	• Accident at TMI-2 • Extensive core damage occurs but coolant flow restored in time to terminate accident progression with core retained in the reactor vessel • Large quantity of H2 generated • H2 combustion event in containment	• 1981 amendment — inert Mark I and II — recombiners — high point vents • 1985 amendment — H2 control system

Table 4-1 Evolution of Knowledge Regarding Risk From Combustible Gases

TIME	EVENT	REGULATORY RESPONSE
1980s/ 1990s	• Severe Accident Research Program • Confirmed ignition limits for variety of H2/air/steam mixtures • Evaluated effectiveness of H2 mitigative systems; example, — igniters work at low H2 concentrations • Established basis for detonability of H2; examples, — possibility of detonation given composition — not a concern for large volume containments • Studied H2 transport and mixing	• no regulations imposed
	• Severe Accident Risks (NUREG-1150) • Other accidents (e.g., Station Blackout (SBO)) also found to contribute to CDF • H2 combustion significant contributor to early containment failure for Mark III and ice condensers during SBO • H2 combustion not a contributor to early failure for large volume containments • H2 and CO contributors to late containment failure	
	• Individual Plant Examination (IPE) Program: Perspectives (NUREG-1560) • Wide range of accident initiators found to contribute to CDF • Per IPEs, H2 combustion — not a contributor at ice condensers because of small SBO contribution — contributor at Mark III because of the high SBO contribution	
	Research (Direct Containment Heating (DCH) Issue Resolution) — Analysis of the challenge to containment integrity from DCH for large dry and ice condenser containments — H2 combustion found to be a challenge to containment integrity for containment integrity for ice condensers during SBO	

During the 1960s and 1970s designers, operators and regulators of LWRs recognized the potential for generating hydrogen by the following mechanisms:

- Metal-water reaction involving metals in the reactor core (cladding and metal structures) and the reactor coolant,

- Radiolytic decomposition of the reactor coolant, and

- Corrosion of metals.

Accidents that could generate significant amounts of hydrogen by these mechanisms were considered to be extremely unlikely however, early regulatory guidance was provided which included consideration of H2 generation. Radiolytic decomposition of the reactor coolant and corrosion of metals are extremely slow processes and can be controlled, by systems such as recombiners with relatively long response times. In addition, in order for a metal-water reaction involving the fuel cladding and the reactor coolant to occur, the core has to be at a high (>1800EF) temperature. At these high temperatures the zircaloy cladding will rapidly oxidize in a steam environment. This oxidation is exothermic and produces combustible hydrogen gas as a reaction product. In a design basis loss of coolant accident (LOCA) the length of time that the core is calculated to be at high temperature prior to ECCS actuation and core reflood is very short and the fraction of the core calculated to be at high temperature is relatively small. Consequently the amount of hydrogen generated is predicted to be relatively small. However, if ECCS is not actuated in a timely manner in this or any other accident sequence, continued oxidation and core degradation can occur.

The WASH-1400 study [1], which was published in 1975, was the first attempt to perform an integrated risk assessment that included accidents in which the reactor core was assumed to melt (i.e., accidents more severe than those considered in the design basis accident (DBA) analysis). Two commercial nuclear power plants (NPPs), namely Surry (a PWR with a subatmospheric containment) and Peach Bottom (a BWR with a Mark I containment) were studied. The results published in WASH-1400 demonstrated that accidents (e.g., transients) other than LOCAs can contribute to the total core damage frequency (CDF). The results also showed that core melt accidents are more important to risk than DBAs.

Significant hydrogen generation was predicted to occur in WASH-1400 as a result of in-vessel clad oxidation during core melt accidents for both NPPs. In addition, the containments were predicted to fail with relatively high conditional probabilities if core melt occurred. Although hydrogen combustion contributed to the high containment failure probabilities reported in WASH-1400 it was not a dominant contributor because other failure mechanisms were considered to be more important at that time. Full core melt accidents were considered in WASH-1400 and core-concrete interactions were modeled including the potential for CO2 generation, which was found to be an important contributor to late overpressurization failure of the containment for some accident sequences. However the reduction of CO2 to combustible CO was not modeled at that time and therefore its impact on late combustion was not determined. Although WASH-1400 was very influential, and it did point out the significance of core melt accidents to risk, it did not impact the regulations related to combustible gas control that were issued during the late 1970s.

The original combustible gas control regulation (10 CFR 50.44) became effective in 1978 with an emphasis on addressing the consequences of only hydrogen generation as a result of the design basis LOCA. This DBA assumed that the initial metal-water reaction would take place over a short period of time early in the LOCA, near the end of the blowdown period and the core refill phase following successful ECCS operation. The duration of the blowdown and core refill phase is on the order of several minutes [Reg Guide 1.7, 1978]. Thus it was felt that the assumption of a 2 minute evolution time for hydrogen (50.44 (d)(1)) from the metal-water reaction, which represents the period of time during which the maximum heat-up occurs, with a constant reaction rate would be conservative for the design of a hydrogen control system. Therefore the limited quantity of hydrogen that had to be addressed in the original regulation resulted in requirements for the installation of recombiners and/or purge systems as discussed in Chapter 3.

The accident at TMI-2 resulted in significant core melting, a large quantity of hydrogen generation (400 kg), and a combustion event in containment. Although the reactor core was severely damaged, coolant injection was restored prior to the core melting through the lower head of the reactor pressure vessel (RPV). Thus no ex-vessel interactions occurred that could release CO during this accident. The accident was, therefore, terminated with the damaged core retained in the RPV. Hydrogen generation occurred as a result of oxidizing approximately 45% of the cladding. The accident had a significant impact on the requirements ultimately imposed by 50.44 and resulted in two amendments to the regulation in 1981 and 1985 (see Table 4-1 and Chapter 3). First, the small volume containments, the Mark I and II BWRs, were required to be inerted, i.e., maintain an oxygen-deficient atmosphere, during power operation. Second, a quantity of hydrogen equivalent to a metal-water reaction of 75% of the clad surrounding the active fuel region was specified in the amendments. This quantity of hydrogen was considered to be representative of a wide range of degraded core accident sequences. In addition, as the TMI-2 accident was terminated, it was assumed that hydrogen was the only combustible gas to be considered and that power was available (i.e., coolant injection was restored) so that any hydrogen control system installed could be designed to use on-site power (i.e., station blackout (SBO) accidents need not be considered on the basis of low probability). The second amendment was restricted in its application only to the “intermediate volume” BWR Mark III and the PWR ice condenser containments.

The requirements imposed in the 1981 and 1985 amendments to 50.44 were intended to address degraded core accidents and reflected our understanding of hydrogen generation and combustion at that time. It was however recognized when 10 CFR 50.44 was amended that we had limited understanding of the behavior of accidents involving severe core damage. The TMI-2 accident, therefore, had a significant impact on research activities sponsored by the NRC and the nuclear industry. Studies related to combustible gas generation, transport, and combustion were an important component of these activities. The objective of the severe accident research program (SARP) sponsored by NRC was to improve our understanding of core melt phenomena and develop improved models to predict the progression of severe accidents.

During the 1980s and 1990s, NRC sponsored research focused both on experimental phenomena and on model development. Experiments were carried out at a variety of scales and under mixture and combustion conditions characteristic of severe accidents in nuclear power plants. Combustion related issues that were studied included:

- combustible gas generation from zircaloy and steel oxidation, core-concrete interactions (CCI), radiolysis, and corrosion.
- Transport and mixing of combustible gases within containment.
- Flammability limits for a range of combustible gas mixtures.
- Combustion pressure-temperature response.
- Diffusion flames and jets.
- Deflagration-to-detonation transitions and detonation limits.
- Mitigation option, including glow plug igniters.

An accurate understanding of the rate and quantity of combustible gas generation is critical for determining the magnitude of the threat posed by combustion. As noted above the 1985 amendment to 50.44 specified a hydrogen source term representative of a degraded core melt accident. An aspect of SARP was therefore directed at improving our understanding of combustible gas generation during degraded core accidents and improving our ability to predict hydrogen generation from zircaloy oxidation during in-vessel core melting. In addition, the severe accident codes were modified to included steel oxidation (not previously modeled) as an additional in-vessel source of hydrogen. SARP also addressed ex-vessel interactions expected during full core melt accidents and greatly improved our understanding of CCI. The importance of other metal constituents in the core debris as sources of combustible gases was modeled in the severe accident codes. Also the production of combustible CO gas from the CO2 released during the interactions of the molten core with limestone concrete was included in the codes.

The importance of ensuring a well mixed atmosphere when combustible gases are released to containment was recognized in the original version of 50.44. If the atmosphere is not mixed stratification of the combustible gases can occur resulting in locally very high concentrations, which can be detonable. Codes were developed to assess the likelihood of achieving a well mixed atmosphere through natural or forced processes for a variety of containment designs. The codes were benchmarked against several experiments.

An understanding of the pressure-temperature response of the containment atmosphere to combustion events (deflagrations and detonations) is essential if the threat to containment integrity is to be accurately determined. SARP therefore focused on improving codes used to predict the pressure-temperature response to combustion events in several different containment designs. These studies confirmed the robustness of large volume and subatmospheric containments in terms of mitigating combustion events, which supported the position adopted in the amendments to 50.44 that did not required a hydrogen control system to be installed in containments with either of these designs. The studies also confirmed the continuing need for combustible gas control (inerting or igniters) in containments with pressure suppression designs.

For some containment designs (e.g., BWRs with Mark III containments) it is possible for standing diffusion flames to form during some core melt accidents. During a transient initiated core melt accident in a BWR with a Mark III containment, hydrogen and steam are released through the tailpipes into the suppression pool. The steam is condensed in the water and a very rich hydrogen mixture is released from the top of the suppression pool into the outer containment. This source of H2 will continue for as long as the oxidation process continues and if it is ignited (by the thermal igniters) could burn as a standing diffusion flame. A number of experiments, significant code development and analyses were performed to address the potential impacts of diffusion flames. The results of this research was incorporated into assessing the effectiveness of the igniter systems.

A detonation produces a dynamic pressure pulse that is much larger than the static pressure loads associated with deflagrations. Detonations are potentially of concern for all containment types. It is therefore important to understand when a detonation might occur (i.e., establish detonation limits for a range of H2, H2O, CO, and CO2mixtures) and when a flame might accelerate and transition into a detonation wave. A significant number of experiments were conducted under SARP to address these issues. A 10% hydrogen concentration was establish as a limit below which a detonation is unlikely to occur.

The 1985 amendment to 50.44 required BWR Mark I and PWR ice condenser containments to be provided with a system capable of controlling H2 from a metal-water reaction of 75% of the cladding. Thermal igniter systems were installed in all of these containment to control this specified quantity of hydrogen. A component of SARP therefore examined the effectiveness of the igniter systems under a variety of conditions.

The results from some of these research activities were incorporated into severe accident codes which were in turn used in a series of studies (e.g., the NUREG-1150 [2] program and the probabilistic risk assessments (PRAs) performed as part of the Individual Plant Examination (IPE) program[3]) to quantify the risk posed by severe accidents for LWRs.

The research, analyses, and studies led to an improved understanding of combustible gas behavior during severe accidents. These findings led to:

- reduced concern for hydrogen combustion in large dry and subatmospheric containment

- confirmation of the need to inert Mark I and II containments

- understanding of the efficacy of igniters in different scenarios at Mark III and ice condenser containments

- mixing induced by igniters

The NUREG-1150 PRAs provide the most thorough PRA treatment of severe accident phenomena to date. Additional insights can be obtained from the industry IPEs, although these studies are often less thorough with respect to considering severe accident phenomena. PRAs typically consider three time regimes for treatment of threats from combustible gasses:

- During the in-vessel core damage process
- At vessel breach or during other major RCS failures (i.e., hot leg), and
- Later in the accident sequence.

Hydrogen generation during the first two time regimes can influence the probability of early containment failure. In a transient accident sequence in BWRs, this hydrogen is released through the safety relief valve (SRV) tailpipes into the suppression pool. In a transient sequence in a PWR, the evolved hydrogen would be released through the power-operated or safety relief valves to the containment. In LOCAs, the hydrogen would be directly released to the containment atmosphere through the break in the reactor coolant system boundary. At the time of vessel breach hydrogen would be released directly to the region below the reactor pressure vessel. Hydrogen evolved during core degradation can also be released to the containment by operator action through the high point vents on the reactor coolant system.

Combustible gas (H2 and CO) generation from CCI during the last time regime influences the probability of late containment. Hydrogen generation however during the first phases (if not ignited early) can also influence the probability of late failure. If combustible gases from all of these sources is allowed to accumulate in the containment, concentrations can exceed the flammable limits and combustible mixtures can form.

The severe accident risk studies carried out in the NUREG-1150 program addressed (through a process of expert elicitation) issues related to hydrogen and carbon monoxide generation and combustion in terms of the impact on containment failure for each of these three time regimes.

Since combustion, and the means to control it, can directly affect the survivability of the containment during a severe accident, it is useful to discuss the implications of the various risk studies individually for different containment types. 50.44 imposes (refer to Chapter 3) one set of requirements for all containments (i.e., hydrogen monitors, recombiners, and purge/repressurization systems), but also mandates different requirements for specific containment types (e.g., inerting for Mark I and II containments, and, in effect, igniters for Mark III and ice condenser containments). The following section therefore also discusses the current hydrogen combustion challenges in terms of the following three groups of containment designs:

- PWRs with large volume and subatmospheric containments,
- BWRs with Mark I and Mark II containments, and
- BWRs with Mark III and PWRs with ice condenser containments.

4.3 Current Challenge to Containments from Combustible Gases

A discussion of the current risk challenges to the different containment types is presented in this section. The containments are grouped as (1) PWR large volume and subatmospheric, (2) Mark I and Mark II, and (3) Mark III and ice condenser containments. The discussion also addresses each of the current requirements in 50.44 in terms of three containment groups

4.3.1 PWR Large Volume and Subatmospheric Containments

Table 4-2 below shows whether or not there are currently any remaining H_2 combustion challenges, in terms of the conditional large early release probability (CLERP), and the conditional large late release probability (CLLRP) to containment integrity (for these two containment designs) for the three time regimes identified earlier in Section 4.2. The CLERP and CLLRP are defined in Chapter 2. The information in the table is based on the results of the research performed during the 1980s and 1990s.

Table 4-2 CLERP and CLLRP from Combustion for PWR Large Volume and Subatmospheric Containments

Containment Design	CLERP Before Vessel Breach	CLERP at Vessel Breach		CLLRP After Vessel Breach
		With RCS at High Pressure	With RCS at Low Pressure	
PWR Large Volume Containments	<<0.1	<0.1	<<0.1	<0.1
PWR Subatmospheric Containments	<<0.1	<0.1	<<0.1	<0.1

Notes:
(14) The results presented in NUREG-1150 (and in the supporting documentation) and in NUREG-1560 were used extensively when constructing this table.
(15) After vessel breach includes up to approximately 24 hours after the onset of core damage.

Need for Severe Accident Hydrogen Control

The above table indicates that hydrogen combustion is not a significant threat to the integrity of large volume and subatmospheric containments for all three time regimes. These containments have very large internal volumes and are predicted to fail at about three times their design pressures. As previously noted in Section 4.1 these containments have significant capacity for withstanding the pressure loads associated with hydrogen deflagration. Detonations of sufficient magnitude to fail containment were judged to have a low probability.

NUREG-1150 assessed the risk of containment failure at Zion, a PWR with a large volume containment, and at Surry, a PWR with a subatmospheric containment for each time regime. For Zion the mean conditional containment failure probability (CCFP) before and at vessel breach was estimated at ~0.01 and the contribution to this low probability from hydrogen combustion was very small. The results for Surry are similar to those predicted for Zion. These mean probability estimates are low but subject to uncertainty. The NUREG-1150 study did develop uncertainty distributions and the 95th percentile for Surry is predicted to be ~0.1. The equivalent number for Zion is ~0.05. The contribution of hydrogen combustion to these two estimates was again predicted to be relatively small. This implies that even when uncertainties are taken into account, hydrogen combustion is not a major cause of containment failure before or at the time of vessel breach for this group of containments.

In addition, the magnitude of the release of radioactive material is predicted to be quite large for these containment failures. Typically, consequence analysis codes only predict the occurrence of acute health effects in the surrounding population when the release fractions of the volatile groups (iodine, cesium, and tellurium) exceed approximately ten percent. NUREG-1150 predicted mean releases in the range of 10 percent for these containment failures with the upper end of the uncertainty distribution extending to approximately 30 percent. These containment failures close to the time of vessel breach, while of relatively low probability, would be classified as “large releases” using the definition described in section 4.1.

IPE results for plants with large volume and subatmospheric containments showed that the conditional probabilities of early containment failure ranged from negligible to about 0.3. The main contributor to the higher probabilities, however, was found to be from sequences leading to high pressure melt ejection, not from hydrogen combustion. Early failure due to over pressurization from hydrogen combustion loads was assessed to be unlikely due to the high pressure capabilities and large volumes of these containment types. Another contributor to the low failure probability was the estimates of the likelihood of a spurious ignition source capable of igniting a hydrogen rich mixture and thus controlling excessive hydrogen buildup.

Although hydrogen combustion does not contribute to the CLERP before or at vessel breach, significant quantities of combustible gases (hydrogen and CO) can accumulate to very large concentrations after vessel breach. The major source of combustible gases in this time frame, in addition to the metal-water reaction, is the core-concrete interaction. Depending on the concrete constituents (limestone or basalt), the core-concrete interaction can be a significant source of carbon dioxide which is subsequently reduced to combustible CO. Combustion events in conjunction with an already existing elevated containment pressure were identified in some IPEs as mechanisms leading to containment failure after vessel breach in individual PWR large volume and subatmospheric containment plants [3]. However, the magnitude of the radionuclide release associated with these containment failures after vessel breach was found to be relatively low (less than 1% of the volatiles released) in the IPEs. Only a small fraction (less than 0.1) of the failures resulted in releases that approach 10%. Therefore, the releases that were predicted to occur in the IPEs after vessel breach (but within 24 hours after the onset of core damage) would not meet the requirements for a large release as defined in section 4.1.

The two amendments to 50.44 required the installation of systems to control hydrogen released during a severe accident for all of the pressure suppression containments. PWRs with large volume and subatmospheric containments were not required to install a system to control hydrogen. Generic Issue-121 [4] addressed the problem of hydrogen control in large volume containments. The resolution of this issue was that hydrogen combustion was not a failure threat for large volume containments and that there was no basis for requiring generic hydrogen control measures, such as igniters, in these plants. The results of the risk studies described above confirm the validity of the resolution of GI-121.

Need for Measuring Hydrogen Concentration

The requirement to measure the hydrogen concentration in containment was imposed in the original version of 50.44. However, a hydrogen control system is not required to mitigate the consequences of a full core melt accident in a large volume or subatmospheric containment, therefore, it is not necessary to measure the hydrogen concentration from the perspective of controlling combustion. The requirement to provide a system to measure the hydrogen concentration in containment does not, therefore, contribute to the risk estimates described above for core melt accidents for these containment designs.

For accident management purposes the hydrogen monitors are used to confirm the amount of core degradation and whether or not an explosive mixture does exists inside containment. Licensees typically define the highest Emergency Action Level, a General Emergency, as a loss of any two barriers and potential loss of the third barrier. Potential loss of a third barrier includes whether or not an explosive mixture exists inside containment. For performing this function the current safety grade monitors with their limited hydrogen concentration range are not the optimum choice. Commercial grade monitors with the ability to monitor a wider range of hydrogen concentration and, preferably, the ability to function under SBO conditions, could be a better solution.

Need for Ensuring Mixed Containment Atmosphere

The requirement to ensure a mixed containment atmosphere was also imposed in the original rule prior to the TMI-2 accident to address the slow evolution of hydrogen from a design basis LOCA accident. Ensuring a well mixed containment atmosphere during a core melt accident is also important because if local pockets of combustible gases accumulate they can form detonable mixtures. However the results of the risk studies noted above indicate that hydrogen combustion is not a significant contributor to CLERP or CLLRP. This statement is true even when uncertainties are considered. This requirement is, therefore, not risk significant for this group of containments.

Need for LOCA Hydrogen Control

The requirement for a hydrogen control system to deal with the slow evolution of hydrogen following a LOCA was also requirement of the original rule. The installation of recombiners and/or vent and purge systems addressed the limited quantity and rate of hydrogen generation that was postulated in the original rule. These systems can only deal with a very limited amount of hydrogen and would be completely overwhelmed by the quantity and rate of hydrogen expected to be evolved during the early stages of a core melt accident in either a large volume or subatmospheric containments. Therefore, these systems are not useful during the three time regimes (identified in Table 4.2) and do not contribute to risk estimates discussed earlier in this section. In addition, in some plants operation of the (backup) purge systems could be problematic in a severe accident situation as it would potentially create a direct path for fission product release outside of containment. When evaluating the need for these systems this potential negative risk impact should be considered.

Need for High-Point Vents

The requirement to install high-point vents was imposed by the first post-TMI amendment to the rule. The vents are actually one of the means by which hydrogen can be introduced into the containment. Design requirements ensure that the potential of the vents as LOCA sources is limited. They were installed to permit venting of non-condensible gases from the reactor coolant system that could potentially impede the operation of the emergency core cooling system, and are therefore more risk significant for ECCS operation than for maintaining containment integrity. The vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting non-condensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus prevent further accident progression. Since continued accident progression can lead to complete core melt and vessel failure, resulting in a threat to containment integrity, the vents do have some mitigative value for reducing the likelihood for early containment failure. However, the risk studies noted above indicate low CLERP both before vessel breach and at the time of vessel breach. Thus the reduction in the likelihood for early failure does not appear to be significant for these containment designs.

Conclusion: *For PWR large volume and subatmospheric containments, H2 combustion does not pose a challenge to containment integrity and therefore, there is no concern for a large release within 24 hours from the onset of core damage. However, the possibility exists for the accumulation of significant quantities of combustible gases (H2 and CO) in the long term (i.e., after several days), which should be considered during implementing accident management strategies.*

4.3.2 BWR Mark I and Mark II Containments

Table 4-3 below shows whether or not there are currently any remaining H_2 combustion challenges to containment integrity (in terms of the CLERP and CLLRP) for these two containment types. The information in the table is again based on the results of the research performed during the 1980s and 1990s.

Table 4-3 CLERP and CLLRP from Combustion for BWR Mark I and Mark II Containments

Containment Design	CLERP Before Vessel Breach	CLERP at Vessel Breach		CLLRP After Vessel Breach
		With RCS at High Pressure	With RCS at Low Pressure	
BWR Mark I	<<0.1	<<0.1	<<0.1	<0.1
BWR Mark	<<0.1	<<0.1	<<0.1	<0.1

Notes:
(1) The results presented in NUREG-1150 (and in the supporting documentation) and in NUREG-1560 were used extensively when constructing this table.
(2) After vessel breach includes up to approximately 24 hours after the onset of core damage.

Need for Severe Accident Hydrogen Control

The above table indicates that the contribution to from H2 combustion is very low during the three time regimes for these containment designs. This is because in the 1981 amendment to 50.44 all BWRs with Mark I and II containments were required to have an inert atmosphere during normal plant operation. Therefore risk studies for plants with Mark I and Mark II containments all include the fact that the containments are inert and, therefore, containment failure and hence a large release due to hydrogen combustion is not possible.

However, given the potentially large concentration of hydrogen that a core damage accident could cause in these plants the likelihood of containment failure from hydrogen combustion would be very high (essentially unity) if the containment were not inert. It has also been determined in risk studies [2-3] that a significant number of these failures can occur in the drywell so that the release would bypass the suppression pool. Releases that bypass the pool tend to be large (i.e., no pool scrubbing) and would, therefore, contribute to the CLERP. It is, therefore, clear that these containment designs should continue to operate with an inert atmosphere in order to meet the numerical guidance defined in Chapter 2.

Hydrogen combustion is prevented in the Mark I and II containments during the early stages of a core melt accident because they are inert. However, both hydrogen and oxygen are generated by the radiolysis of coolant solutions inside and outside the reactor coolant system due to absorption of the radiation emitted by the released fission products. The rate of production of these gases depends on the amount and quality of radiation energy absorbed in the specific coolant solutions used and the net yield of gases generated from the solutions due to the absorbed radiation energy. The yield is affected by numerous factors such as coolant flow rates, chemical additives and impurities in the coolant, temperature, etc. Regulatory Guide 1.7 [5] recommends assumptions and values of the fraction of fission product energy absorbed by the coolant and the hydrogen and oxygen yield rates as a function of the absorbed energy that are acceptable for calculating the production of gases from radiolysis. Reg Guide 1.7 also recommends an oxygen concentration limit of 5 v/o if combustion is to be prevented assuming a hydrogen concentration of $ 6 v/o.

While the evolution of gases from radiolysis takes place at a much lower rate compared to the zirconium-water reaction, a combustible mixture could eventually form late (i.e., on the order of days after the onset of core damage) in the accident sequence from the evolution of oxygen. This implies that potential for hydrogen combustion in the long term should be considered when implementing severe accident management strategies.

Need for Measuring Hydrogen Concentration

The requirement to measure the hydrogen concentration in containment was imposed in the original version of 50.44. However, during the first two time regimes of a full core meltdown accident (identified in Table 4.3) it is not necessary to measure the hydrogen concentration in BWR Mark I and II containments because they are inert and no actions would be taken based on this measurement. The requirement to provide a system to measure the hydrogen concentration in containment is, therefore, not risk significant during the early stages of a core-melt accident.

In BWR Mark I and II containments, hydrogen (and oxygen) monitoring can have value late in an accident sequence when severe accident management considerations apply. Because hydrogen combustion is unlikely in the early stages due to inerting, the hydrogen monitors can provide an accurate indication of core damage in later phases of the accident. For combustion control, oxygen monitoring is more important than hydrogen monitoring for these containment designs. One source of oxygen late in the accident sequence is from the slowly evolving source of radiolysis that can pose a combustion threat, however this source can be controlled with recombiners. If hydrogen and oxygen monitors are unavailable, e.g. during a SBO, so that the concentrations can not be determined, and other indicators show evidence of core damage then current plant procedures recommend containment venting irrespective of the offsite radioactivity release rate.

For BWR Mark I and II containments, hydrogen concentration appears extensively in the emergency procedure guidelines (EPGs)/severe accident guidelines (SAGs), including as an entry condition. As such the need for measuring the H2 concentration should be assessed in the context of supporting the EPGs/SAGs.

Need for Ensuring Mixed Containment Atmosphere

The requirement to ensure a mixed containment atmosphere was also imposed in the original rule prior to the TMI-2 accident to address the slow evolution of hydrogen from a design basis LOCA accident. Ensuring a well mixed containment atmosphere during a core melt accident is also important because if local pockets of combustible gases accumulate they can form detonable mixtures. However, BWR Mark I and II containments are inert and therefore combustion is prevented so that the possibility of forming local pockets of combustible gases is not of concern. This requirement is therefore not relevant to these containment designs.

Need for LOCA Hydrogen Control

The requirement for a hydrogen control system to deal with the slow evolution of hydrogen following a LOCA was a requirement of the original rule. The installation of recombiners and/or vent and purge systems addressed the limited quantity and rate of hydrogen generation that was postulated in the original rule. These systems are not needed during the three time regimes (refer to Table 4.3) of a core melt accident in BWRs with Mark I and II containments because the atmospheres are inert.

Need for High-Point Vents

The requirement to install high-point vents was imposed by the first post-TMI amendment to the rule. The vents are actually one of the means by which hydrogen can be introduced into the containment. Design requirements ensure that the potential of the vents as LOCA sources is limited. They were installed to permit venting of non-condensible gases from the reactor coolant system that could potentially impede the operation of the emergency core cooling system, and are therefore more risk significant for ECCS operation than for maintaining containment integrity. The vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting non-condensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus prevent further accident progression. Since continued accident progression can lead to complete core melt and vessel failure, resulting in a threat to containment integrity, the vents do have some mitigative value for reducing the likelihood for early containment failure. However, as the BWR Mark I and II containments are inert all combustion is prevented so that there is no difference between the CLERP before vessel breach and after vessel breach in terms of the threat from combustion. This is not however true if other modes of containment failure are taken into account. There is a significant difference between the CLRP for degraded core and for full core melt accidents when all modes of containment are taken into account.

Conclusion: *For BWR Mark I and Mark II containments, combustion is not a challenge to containment integrity solely because of the inert atmosphere. However, the possibility exists for oxygen generation, and therefore, a combustion challenge to containment integrity in the long term (i.e., after several days), which should be considered during implementation of accident management strategies.*

4.3.3 BWR Mark III and PWR Ice Condenser Containments

Table 4-4 below shows whether or not there are currently any remaining H_2 combustion challenges to containment integrity (in terms of the CLERP and CLLRP) for these two containment types. The information in the table is based on the results of the research performed during the 1980s and 1990s.

Table 4-4 CLERP and CLLRP from Combustion for Mark III and Ice Condenser Containments

Containment Design	CLERP Before Vessel Breach	CLERP at Vessel Breach		CLLRP After Vessel Breach
		With RCS at High Pressure	With RCS at Low Pressure	
Mark III with Igniters Operating	<<0.1	>0.1	<0.1	<0.1
Mark III without Igniters Operating	<0.1	>0.1	>0.1	<0.1
Ice Condensers with Igniters Operating	<<0.1	<0.1	<0.1	<0.1
Ice Condensers without Igniters Operating	>0.1	>0.1	>0.1	>0.1

Notes:
(1) The results presented in NUREG-1150 (and in the supporting documentation), NUREG-1560, and NUREG/CR-6427 were used extensively when constructing the table.
(2) After vessel breach includes up to approximately 24 hours after the onset of core damage.

Need for Severe Accident Hydrogen Control

The 1985 amendment to 50.44 required all BWR Mark III and PWR ice condenser containments to install systems to control hydrogen. Therefore for plants with these containment types existing PRA analyses include the igniter systems in the plant model. Nevertheless, hydrogen combustion was still found to be a significant contributor (as indicated in the above table) to early containment failure, and hence the CLERP, in some of the analyses, mainly from station blackout (SBO) sequences. This result is not unexpected because the amendments to 10 CFR 50.44 were written to mitigate terminated accidents (like TMI-2) in which the reactor core is damaged but retained in the RCS and for which power to operate the igniter system is assumed to be available. However, PRAs consider a wider range of accidents (including full-core events) in which the core melts through the RPV. Accidents in which power is not available to operate either the igniter system or the air return fans (ARF), are also modeled in PRAs.

The risk results for Grand Gulf reporting in NUREG-1150 were obtained from a more detailed report [6]. This report (NUREG/CR-4551, Volume 6) presents results that show (if the igniters are operating) a mean conditional probability of containment failure prior to vessel breach in the range of 0.01 to 0.02. Uncertainty distributions about these mean failure probabilities were not displayed in NUREG/CR-4551, Volume 6, but, based on other uncertainty distributions presented in the report, one could conclude that the 95th percentile would be at about 0.1. However, containment failures before vessel breach do not result in a large release for BWR Mark III containments. This is because the releases are scrubbed by the suppression pool. The CLERP is, therefore, significantly less than 0.1 if the igniters are operating.

NUREG/CR-4551 shows that if the igniters are not operating then the mean conditional probability of containment failure prior to vessel breach is about 0.1 to 0.2. The 95th percentile of the uncertainty distribution would be expected to increase these probability estimates to close to unity. However, even though there is a relatively high probability of containment failure, the releases may not be large because the drywell is predicted to remain intact and the releases would be scrubbed by the suppression pool. The CLERP prior to vessel breach is, therefore likely to be less than 0.1 even if the igniters are not operating however, this result is subject to uncertainty.

Predicting the conditional probability of containment failure at the time of vessel breach was found (in NUREG/CR-4551, Volume 6) to be uncertain and dependent on the RCS pressure and on whether or not the igniters were operating. If the RCS is at high pressure, the conditional probability of containment failure was predicted to be approximately 0.5 even with the igniters operating. The 95th percentile of the probability distribution is essentially unity. In addition, approximately 50 percent of these containment failures also resulted in simultaneous failure of the drywell, which leads to early suppression pool bypass and relatively large releases. Thus, if the RCS is at high pressure at the time of vessel breach, the CLERP is greater than 0.1 whether or not the igniters are operating.

If the RCS is depressurized at the time of vessel breach and the igniters are not operating, NUREG/CR-4551, Volume 6, reports conditional probabilities of containment failure greater than 0.5 with a 95th percentile failure probability close to unity. Simultaneous failure of the drywell was also predicted for about 50% of these failures at the time of vessel breach. Failure of the drywell leads to pool bypass (i.e., no pool scrubbing) and, hence, large releases. Thus, if the RCS is at low pressure at the time of vessel breach, the CLERP is greater than 0.1 if the igniters are not operating.

The potential for containment failure caused by hydrogen combustion at the time of vessel breach when the RCS is at low pressure and the igniters are operating is not directly assessed in NUREG/CR-4551, Volume 6. However, the conditions prior to vessel breach should be applicable to this situation because the RCS is depressurized and none of the issues associated with HPME would occur. It is, therefore, reasonable to assume that if the RCS is at low pressure at the time of vessel breach, and if the igniters are operating, then the CLERP should be less than 0.1.

After vessel breach (but within 24 hours of the onset of core damage) NUREG/CR-4551, Volume 6, reports conditional probabilities of containment failure from about 0.1 to 0.3 caused by hydrogen combustion and gradual overpressurization. Again, distributions were not provided but the 95th percentiles would be expected to approach unity. However, even though there is a high probability of containment failure, most of the release would be scrubbed by the suppression pool. The CLLRP after vessel breach is, therefore, less than 0.1 because of pool scrubbing.

The NUREG-1150 and IPE results for Sequoyah show that if the igniters and the ARFs are operating the conditional probability of containment failure and, hence, the CLERP, is significantly less than 0.1 for all three time regimes independent of the RCS pressure (i.e., for all accidents except SBO). Therefore if the igniters and ARFs are operating hydrogen combustion is not a challenge to containment. The IPE results also indicated that even if the igniters and ARFs are not operating (i.e., for SBO sequences etc.) then hydrogen combustion is also not a challenge to the integrity of ice condenser containments. However, this is not substantiated by results in NUREG-1150 and NUREG/CR-6427.

The NUREG-1150 results indicate a mean conditional containment failure probability (CLERP) of just over 0.1 for SBO sequences, and these early failures were predicted to result in large releases. A CLERP of just over 0.1 is borderline in terms of remaining a challenge based on the guidelines in the framework document. In addition, the uncertainty associated with predicting the CCFP (and, hence, CLERP) has a very skewed distribution (reported in NUREG-1150) with a significant density of observations at the 95th percentile. This uncertainty distribution appears to be supported by recent work, documented in NUREG/CR-6427 [7] which indicates that hydrogen combustion does pose a very severe challenge to containment integrity if the igniters are not operating. Calculations made with the CONTAIN code indicated that no ice condenser plant is inherently robust to hydrogen combustion events in a SBO accident. If igniters are not available and other ignition sources are absent, large amounts of hydrogen can accumulate in the containment prior to vessel breach in some accident sequences. The combustion of this hydrogen can greatly augment DCH loads and, in fact, combustion of this hydrogen by itself can threaten containment. Furthermore, the ice condenser cannot mitigate this component of the containment loading to any great extent. Hydrogen deflagration and, possibly, detonation, in the upper dome are more credible in ice condenser plants than in large dry containments. The initial conditions calculated by CONTAIN for SBO scenarios indicate molar hydrogen concentrations of 14%-18% if ignition sources are absent. These calculations are based on hydrogen production calculated by SCDAP/RELAP5 in which in-vessel zirconium oxidation was predicted to be 58%. This value, it should be noted, is less than the 75% clad oxidation postulated for compliance with 10 CFR 50.44. Quantification of containment event trees in NUREG/CR-6427 showed that for both slow and fast station blackout scenarios the conditional probability of containment failure, and, hence, CLERP, due to hydrogen combustion events in the Sequoyah plant is over 0.97 and ranges from 0.22 to 0.95 at other ice condenser containment plants. In the light of these new results hydrogen combustion therefore remains a challenge to the containment integrity of ice condensers in accident scenarios where the igniters are not available.

Need for Measuring Hydrogen Concentration

The requirement to measure the hydrogen concentration in containment was imposed in the original version of 50.44. However, during the three time regimes identified in Table 4.4 it is not necessary to measure the hydrogen concentration in BWR Mark III or PWR ice condenser containments because the igniter systems are actuated based on high pressure. The requirement to provide a system to measure the hydrogen concentration in containment is therefore not risk significant in terms of dealing with the combustion threat during these time regimes of a core melt accident (except for those conditions where the igniters are not operable, e.g., SBO).

For BWR Mark III containments, hydrogen concentration appears extensively in the EPGs/SAGs, including as an entry condition. As such the need for measuring the H2 concentration should be assessed in the context of supporting the EPGs/SAGs.

Need for Ensuring Mixed Containment Atmosphere

The requirement to ensure a mixed containment atmosphere was also imposed in the original rule prior to the TMI-2 accident to address the slow evolution of hydrogen from a design basis LOCA accident. Ensuring a well mixed containment atmosphere during a core melt accident is also important because if local pockets of combustible gases accumulate they can form detonable mixtures. The above results indicate that this an important issue for these containment designs and therefore this requirement is risk significant.

Need for LOCA Hydrogen Control

The requirement for a hydrogen control system to deal with the slow evolution of hydrogen following a LOCA was the third high level requirement of the original rule. The installation of recombiners and/or vent and purge systems addressed the limited quantity and rate of hydrogen generation that was postulated in the original rule. These systems would be completely overwhelmed by the quantity and rate of hydrogen expected to be evolved during the three time regimes identified in Table 4.4. Therefore these systems are not useful during these time regimes and consequently are not risk significant.

Need for High-Point Vents

The requirement to install high-point vents was imposed by the first post-TMI amendment to the rule. The vents are actually one of the means by which hydrogen can be introduced into the containment. Design requirements ensure that the potential of the vents as LOCA sources is limited. They were installed to permit venting of non-condensible gases from the reactor coolant system that could potentially impede the operation of the emergency core cooling system, and are therefore more risk significant for ECCS operation than for maintaining containment integrity. The vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting non-condensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus prevent further accident progression. Since continued accident progression can lead to complete core melt and vessel failure, resulting in a threat to containment integrity, the vents do have mitigative value for reducing the likelihood for early containment failure. The above results indicate that there is a significant difference between the CLRP for degraded core and for full core melt accidents for these containment designs. In particular the risk significant issue associated with high pressure core melt accidents at vessel breach for BWR Mark III containments would be avoided if the damage core could be retained in the reactor vessel.

Conclusion: *For BWR Mark III and PWR ice condenser containments, hydrogen combustion is not a challenge to containment integrity solely when igniters are available and operable, unless (for Mark III containments) the RCS is at high pressure when the core debris melts though the reactor vessel. Under these circumstances, even with the igniters operating hydrogen combustion remains a threat to containment integrity.*

4.4 Summary

Research and risk studies related to combustible gas control suggest that a risk-informed 10 CFR 50.44 should address the following:

- All accident types including full core melt (i.e., the core melts through the reactor vessel) accidents that result in a large release of radionuclides to the environment.

- The extent of the challenge to containment integrity depends on the rate and quantity of combustible gases released. A realistic combustible gas source term should, therefore, include combustible gases generated and released to containment from:

— in-vessel metal-water reactions
— reactor vessel blow down
— ex-vessel core-concrete interactions

- Combustible gas control is needed for some containment types during a core melt accident sequence prior to, during, and after vessel failure (up to approximately 24 hours after the onset of core damage).

The results of research and risk studies related to combustible gas control have been compared to the numerical guidelines in the framework document to determine the risk significance of the requirements in 50.44. The following observations were derived from this comparison:

1) Some of the requirements are risk significant for some containment types:
   S inerting Mark I and II containments
   S providing severe accident hydrogen control for Mark III and ice condenser containments
   S ensuring a mixed containment atmosphere for Mark III and ice condenser containments

2) One requirement is not risk significant for any containment type:
   S LOCA hydrogen control

3) Some requirements may need to be enhanced:
   S ensuring severe accident hydrogen control for Mark III and ice condenser containments for all risk significant accident sequences
   S coupling the requirement for ensuring a well mix containment atmosphere to the requirement for severe accident hydrogen control for Mark III and ice condenser containments

4) One requirement is important to safety but not related to the concern being addressed by the rule:
   S the need for high point vents

These perspectives have been integrated with the other aspects of the framework to provide a risk-informed alternative. This work is described in Sections 5 and 6.

4.5 References

1. US NRC, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

2. US NRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

3. US NRC, "Individual Plant Examination Program Perspectives Report," NUREG-1560

4. US NRC, "Issue 121: Hydrogen Control for Large, Dry PWR Containments (Rev.2)," NUREGs/SR-0933.

5. US NRC, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Regulatory Guide 1.7, Revision 2, November 1978.

6. T.D. Brown, et al., "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," NUREG/CR-4551, Volume 6, Revision 1, Part 1, December 1990.

7. M.M. Pilch et al., "Assessment of the DCH Issue for Plants with Ice Condenser Containments," NUREG/CR-6427, February 2000.

5. POTENTIAL RISK-INFORMED OPTIONS

As discussed in Section 2.2, once the concern associated with a regulation selected as a candidate for the risk-informing process is clearly understood, the two approaches shown in Figure 2.5 are followed. Both approaches have the same overall objective which is to develop risk-informed options for dealing with the identified concern. The identified concern, in this instance, is the threat from hydrogen combustion. The risk significance of hydrogen combustion and the associated needs for controlling it are discussed in detail in Chapter 4.

As the discussion in Chapter 4 indicates, combustible gas control related research and risk studies suggest that any risk-informed alternatives to the current 10 CFR 50.44 should account for the following:

- The combustible gas threat to containment integrity is dominated by accidents that develop to a full core melt with reactor vessel failure
- The rate and quantity of combustible gas released, i.e., the combustible gas source term, considered for accident analysis should be based on realistic calculations which cover all phases of the accident (in-vessel interactions, reactor vessel blowdown, and ex-vessel sources including core-concrete interactions)

The first approach in Figure 2.5 starts from the current set of regulations in Part 50 that address the concern and attempts to develop options based on risk-informing the requirements laid out in the current regulation. The second approach takes a fresh start at developing alternative risk-informed options for addressing the defined concern but without recourse to the existing body of regulations. Although these approaches have different starting points, they should lead in the end to a similar outcome, i.e., risk-informed requirements for dealing with combustible gas concerns.

The first approach is applied to 10 CFR 50.44 by assessing each of the high level requirements of this regulation for their risk significance and assessing the overall regulation for completeness in terms of risk. With this approach options are developed in which individual current requirements may be modified, deleted or left unchanged, and some additional requirements may be added. These options are called “revised” risk-informed options, and are discussed in Section 5.1.

In the second approach, the concern dealt with in 10 CFR 50.44 and analyzed in Chapter 3, i.e, the threat to containment integrity from combustible gas deflagration or detonation, is addressed via options developed by systematically applying the defense-in-depth strategies of the Framework, without regard to the requirements in the current 10 CFR 50.44. The options developed from this “clean sheet” approach are termed “alternate” risk-informed options, and are discussed in Section 5.2. The options developed in Section 5.2 below can each individually replace the current 10 CFR 50.44 regulation, while the options developed in Section 5.1 need to be combined in order to completely address the combustible gas concern. Viable options developed using the both approaches are offered as a comprehensive alternative to the existing 50.44 rule, as discussed in Chapter 6.

As discussed in Section 1.2, this work is intended to demonstrate the feasibility of risk-informed changes to 10 CFR 50.44. If the Commission approves going to rulemaking, additional analyses will be required.

5.1 Revised Risk-Informed Options

Recalling Figure 2.5, the approach based on modifying existing requirements consists of the following five steps:

1. Identify and describe the current requirements
2. Identify and describe related regulations and implementing documents
3. Identify and describe industry implementation of the requirements
4. Determine risk significance of requirements and implementation
5. Identify and describe risk-informed options

The current requirements, imposed by the original rule and two subsequent amendments, can be summarized at a high level as consisting of three analytical requirements and six physical requirements. The analytical requirements, which address the types of accidents examined as well as the sources and amounts of combustible gas to be considered, provide the background and context in which the physical requirements are applied.

The three analytical requirements of the current rule are the following:

- Accidents examined:
    - S postulated LOCA (for all reactors, as specified in the original rule), and
    - S degraded core accidents (for some containment types, as specified in the amendments).
- Sources of combustible gas:
    - S metal-water reaction between the zirconium cladding and the reactor coolant,
    - S radiolytic decomposition of the coolant, and
    - S corrosion of metals.
- Amount of combustible gas:
    - S 5% clad metal/water reaction for all reactors, and
    - S 75% clad metal/water reaction for some containment types.

The six high level physical requirements imposed by the current rule are the following:

- measure the concentration of hydrogen in the containment
- insure a mixed atmosphere in the containment
- control combustible gas concentrations in containment following a postulated LOCA.
- install high point vents on all reactors.
- inert the atmosphere in BWR Mark I and Mark II containments
- provide a hydrogen control system for BWRs with Mark III containments and PWRs with ice condenser containments justified by a suitable program of experiment and analysis.

For each of the high level requirements the five steps listed above were carried out in the identification of each of the options discussed below. The first three steps were discussed in detail in Chapter 3. The risk significance of the combustible gas threat was presented in Chapter 4. Risk-informed options are identified below.

5.1.1 Potential Changes to Analytical Requirements

Risk-informed options for the three analytical requirements are developed below.

5.1.1.1 Requirement: Postulated Loss-of-Coolant Accident and Degraded Core Accident

Requirement

10CFR 50.44(a) states the following:

> *"Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding....include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident..."*

10CFR 50.44(c)(3)(vi)(A) and (c)(3)(vi)(B)(3) states the following:

> *"...for a boiling light-water nuclear power reactor with a Mark III type of containment or for a pressurized light-water nuclear power reactor with an ice condenser type of containment .... Use accident scenarios....resulting in a degraded core."*

Evaluation

As noted in Chapter 3, experience and experiment have shown that during accidents involving core damage sufficient quantities of combustible gases can be evolved to pose a threat for some containments. The most significant risk appears to come from full core melt accidents, which include in-vessel clad metal/water reaction, potentially large quantities of hydrogen entering the containment at vessel failure, and the possibility of core-concrete interaction as the accident continues. On the other hand, design basis LOCA accidents, which involve only minor clad oxidation and in which the reactor vessel and containment does not fail, are not contributors to risk.

Therefore, the emphasis of the current requirement on consideration of postulated loss-of-coolant accidents for all reactors is no longer proper from a risk-informed perspective, and the consideration of degraded core accidents for reactors housed in BWR Mark III or PWR ice condenser containments should be expanded in two ways:

- degraded core accidents should be expanded to full core melt accidents, and
- the emphasis on core melt accidents should extend to all containment types

Proposed Change

- Modify 50.44 (a) by deleting the words "postulated loss-of-coolant accident (LOCA)" in the rule and replacing them by "core melt accident", and add suitable words to indicate that the combustible gas generation during the entire accident progression should be accounted for.

5.1.1.2 Requirement 50.44: Combustible Gas Sources

Requirement

10CFRR 50.44 (a) states the following:

> *"....include means for control of hydrogen gas that may be generated, ... by (1) metal-water reaction involving the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the reactor coolant, and (3) corrosion of metals."*

Evaluation

The only combustible gas mentioned in the current requirement is hydrogen. While hydrogen generated by clad metal/water reaction is the principal source of combustible gas in core melt accidents, a significant amount of carbon monoxide can also be generated from core-concrete interactions (CCI) when the accident has progressed to the ex-vessel stage, given limestone based concrete and a dry reactor cavity.

In addition, for the in-vessel reaction there may be other oxidation sources present besides the clad metal, such as the channel boxes in BWRs.

Proposed Change

- Modify 50.44 (a) by adding suitable words to the rule to indicate hydrogen generation from metal-water reaction also involves other sources of metal in the core besides the fuel cladding, and that carbon monoxide evolved from core-concrete interaction is another combustible gas that should be considered. Therefore the wording should indicate that combustible gases may be generated following a full core melt accident by:

  (1) Metal-water reaction involving, principally, the fuel cladding and the reactor coolant but also including other sources in the core such as channel boxes in BWRs,
  (2) Radiolytic decomposition of the reactor coolant,
  (3) Corrosion of metals, and
  (4) Metal reactions during core-concrete interaction.

5.1.1.3 Requirement 50.44: Combustible Gas Source Term

Requirement

For reactors with BWR Mark III or PWR ice condenser containments, 10CFR 50.44(c)(3)(iv)(A) states the following:

> *"Each ...reactor with a Mark III type of containment and .... with an ice condenser type of containment....The hydrogen control system much be capable of handling....hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding..."*

10CFR 50.44 (c)(3)(v)(B) states the following:

> *"The amount of hydrogen to be considered is equivalent to that generated from a metal-water reactions involving 75% of the fuel cladding...."*

10CFR 50.44 (c)(3)(vi)(B)(1) states the following:

> *"...hydrogen resulting from the metal-water reactions of up to and including 75% of the fuel cladding..."*

For all reactors, 10CFR 50.44 (d) states the following:

> *"...the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water)....shall be assumed either to be* [5%]*or ...from reaction of all the metal ... to a depth of 0.00023 inch, whichever amount is greater..."*

Evaluation

The amount of hydrogen generated in a 75% metal/water reaction specified in 10 CFR 50.44 (c)(3)(iv), (v) and (vi) for Mark III and ice condenser containments focuses on core degradation. However, it is presently not conclusively demonstrated that this is a realistic amount of combustible gas for risk-significant accident scenarios in all reactor types. Also, the rate at which hydrogen is assumed to enter the containment in the analyses required under the present 10 CFR 50.44 (c)(3)(v)(B) is representative of a degraded core accident and not a full core melt accident.

The amount of combustible gas proposed in 10 CFR 50.44 (d) for consideration by all reactors is based on a postulated design basis LOCA. As noted in Chapter 3, understanding of the combustion threat has matured to the realization that this small amount of hydrogen is not the principal contributor to risk, and is dwarfed by the comparatively large, quickly evolving amounts of combustible gas generated in a core melt accident.

Proposed Change

- Modify the appropriate sections of 50.44 by calling for the use a specified combustible gas source term, generated by the NRC staff, with this risk-informed requirement. A series of specific source terms, tailored to containment types and accident scenarios would be described in a Regulatory Guide. The source terms would account for all phases of an accident (as discussed in 5.1.1.2 above), including in-vessel metal/water reaction, vessel blowdown, and ex-vessel core/concrete interaction in the later stages of the accident.

5.1.2 Potential Changes to Physical Requirements

The six high level physical requirements of the current rule are the following:

- measuring the concentration of hydrogen in the containment
- insuring a mixed atmosphere in the containment
- controlling combustible gas concentrations in containment following a postulated LOCA
- installation of high point vents on all reactors
- an inerted atmosphere for Mark I and Mark II containments
- a hydrogen control system for BWRs with Mark III containments and PWRs with ice condenser containments justified by a suitable program of experiment and analysis.

5.1.2.1 Requirement 50.44 : Measure H2 Concentration

Requirement

10 CFR 50.44 (b)(1) states the following:

> *"...(b) Each boiling or pressurized light-water reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with the capability for ... (1) Measuring the hydrogen concentration in the containment..."*

Licensees have met this requirement by installing continuous safety-grade monitors.

Implementing guidance is provided by the following:

Regulatory Guide 1.7 [1] recommends that systems to measure combustible gases in containment should meet the requirements for an engineered safety feature.

Regulatory Guide 1.97 [2] treats hydrogen monitors as redundant, safety-grade Class 1E electrical equipment. Consequently, they are subject to the procurement requirements of 10 CFR 21 and the quality assurance requirements of Appendix B to Part 50.

Regulatory Guide 1.101, Revision 3 [3] endorses NEI-NESP-007, Revision 2 which states that a General Emergency is a loss of any two barriers and potential loss of the third barrier. Potential loss of a third barrier includes whether or not an explosive mixture exists inside containment. The continuous hydrogen monitors are credited for making this determination.

Evaluation

The current requirement 50.44(b)(1) to provide a measurement capability is specifically stated in the context of a postulated loss-of-coolant-accident (LOCA). The intent of the current regulation is to provide a measurement capability to alert the operators to initiate and verify hydrogen control measures (systems such as recombiners, purge systems, etc.) to keep the hydrogen concentration below the 4% flammability limit (or the oxygen concentration below the 5% limit in BWRs) following a design basis LOCA. As stated earlier, the DBA LOCA is not a risk-significant accident, hence the measures used to control the amounts of hydrogen generated in this accident and the monitoring used to actuate these measures are also not risk-significant.

In terms of the risk-significant core melt accidents, 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 spurious 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.

In BWR Mark I and II containments, hydrogen (and oxygen) monitoring can have value late in an accident sequence when severe accident management considerations apply. Because hydrogen combustion is unlikely in the early stages due to inerting, the hydrogen monitors can provide an accurate indication of core damage in later phases of the accident. For combustion control, oxygen monitoring is more important than hydrogen monitoring for these containment designs. One source of oxygen late in the accident sequence is from the slowly evolving source of radiolysis that can pose a combustion threat, however this source can be controlled with recombiners. If hydrogen and oxygen monitors are unavailable, e.g. during a SBO, so that the concentrations can not be determined, and other indicators show evidence of core damage then current plant procedures recommend containment venting irrespective of the offsite radioactivity release rate.

For accident management purposes the hydrogen monitors are used to confirm the amount of core degradation and whether or not an explosive mixture does exists inside containment. Licensees typically define the highest Emergency Action Level, a General Emergency, as a loss of any two barriers and potential loss of the third barrier. Potential loss of a third barrier includes whether or not an explosive mixture exists inside containment. For performing this function the current safety grade monitors with their limited hydrogen concentration range are not the optimum choice. Commercial grade monitors with the ability to monitor a wider range of hydrogen concentration and, preferably, the ability to function under SBO conditions, could be a better solution.

NUREG-0737 [4], Item II.F.1, Attachment 6 requires that all plants provide a continuous indication of hydrogen concentration in the containment for accident monitoring. The post-TMI requirement also imposed the design and quality criteria of Regulatory Guide 1.97 and required that the hydrogen monitors be included in a plant's technical specifications. The continuous hydrogen monitors are also credited in meeting 10 CFR 50.47 (b)(9) and Appendix E to Part 50 for assessing and monitoring offsite consequences. The staff recently granted an exemption to NUREG-0737, Item II.B.3, postaccident sampling of containment atmosphere hydrogen concentration, because the continuous hydrogen monitors were an acceptable alternative.

It is also important to note that this requirement is redundant to NUREG-0737, Item II.F.1, Attachment 6, 50.47, and Part 50 Appendix E.

Therefore, since the need is the capability of establishing hydrogen concentration levels, under degraded core conditions, for long term accident management, and since the hydrogen monitors are not necessary to meet any of the defense-in-depth elements (as listed in Section 2), the recommendation is to delete the requirement from 10 CFR 50.44, and that the guidance provided by NUREG-0737, Item II.F.1, Attachment 6 and RG 1.97, Revision 2 be revised to allow commercial-grade hydrogen monitors with a range from 0 to an amount of hydrogen generated from 100% of the active fuel clad reacting with water, or 30%, which ever is less. In addition, it would be desirable that the monitors be available during SBO sequences and be able to survive the effects of a degraded core accident including those initiated by external events. It is not recommended that design and quality criteria be changed for the containment high range radiation monitors or the oxygen monitors based on the above risk insights.

Proposed Change

- Eliminate the requirement for measuring hydrogen concentration by removing 50.44 (b)(1), redundant to NUREG-0737, Item II.F.1, Attachment 6, 50.47, and Part 50 Appendix E.

- Given the need for the capability of establishing hydrogen concentration levels, under degraded core conditions, for long term accident management, recommend that related regulations be revised to remove continuous measuring and safety-grade requirements and call for an increased measurement range.

5.1.2.2 Requirement 50.44: Insure a Mixed Containment Atmosphere

Requirement

10 CFR 50.44 (b)(2) states the following:

> *"...(b) Each boiling or pressurized light-water nuclear reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with the capability for ... (2) Insuring a mixed atmosphere in containment..."*

Licensees have met this requirement by using other equipment already in the plant to perform this function. This equipment varies according to containment type, but generally involves fan systems and/or containment spray systems. In ice condenser containments, for example, both the safety-grade Air Return Fans (ARFs) and the sprays are used. These systems serve a number of safety functions including containment heat removal. The systems must comply with the General Design Criteria (GDC) 41, 42, and 43 of Appendix A of Part 50.

Evaluation

Although the requirement of 50.44 (b)(2) was promulgated to address post-LOCA hydrogen accumulation, a mixed containment atmosphere is beneficial for core melt accident conditions also since a well mixed atmosphere prevents local accumulation of combustible or detonable gas concentrations which could threaten containment integrity. The risk significance of a well mixed atmosphere varies depending on containment type.

For the smaller volume containments, i.e., the BWR Mark I and Mark IIs, the risk significance of a mixed atmosphere could be very high since the small volumes and confined spaces of these containments could easily lead to build-up of significant hydrogen concentrations, given a core melt accident. However, another part of 50.44 requires the BWR Mark I and Mark II containments to be inerted. Inerting is an effective hydrogen control system for all risk-significant degraded core and full core melt accidents in these containments. Therefore, because of the inerted nature of these containments, the risk-significance of keeping the atmosphere mixed to prevent hydrogen combustion is actually quite low for BWR Mark I and Mark IIs.

For the intermediate containments, i.e., the BWR Mark IIIs and the PWR ice condensers, the risk-significance of a mixed atmosphere could again be very high for reasons similar to those mentioned above for BWR Mark I and Mark II containments; i.e., relatively small volumes and geometries susceptible to hydrogen pocketing. However, as required by other parts of 50.44, these containments also have hydrogen control systems, i.e, deliberate ignition systems, effective for mitigating the hydrogen threat from degraded and full core melt accidents. If these deliberate ignition systems operate for all risk-significant sequences, then the risk-significance of ensuring a mixed atmosphere is also relatively low for BWR Mark III and PWR ice condenser containments.

The large volume and relatively open geometry of large dry containments makes the accumulation of high concentrations of hydrogen unlikely in these plants. This open geometry also supports atmospheric mixing brought about by the phenomena of the core melt accident itself, such as blowdowns from pipe ruptures or reactor vessel failure. During the IPE process licensees with large dry containments were asked to evaluate their containments for susceptibility to local hydrogen concentration. These evaluations indicated that either no possibility for local hydrogen accumulation was identified, or that locations where accumulations could occur did not contain equipment whose failure (as a result of hydrogen combustion) would affect plant risk. Therefore, for large dry containments the risk-significance of ensuring a mixed atmosphere is relatively low even with no dedicated hydrogen control systems in place, as long as the open geometry is maintained.

Although the risk-significance of this requirement is relatively low, provided hydrogen control systems are in place for risk-significant accident sequences in all BWR containments and in PWR ice condenser containment, it is retained for defense-in-depth reasons. As discussed in the framework document and in Chapter 2 of the present report, one of the considerations in the defense-in-depth approach for applying the four strategies of Figure 2-1 is that the intent of the General Design Criteria (GDC) of Appendix A to 10 CFR 50 is maintained. GDC 50 addresses the containment design basis and notes that the containment and its compartments shall accommodate, with sufficient margin, the effects of potential energy sources including those of 50.44, i.e., energy from metal-water and other chemical reactions. Retaining this requirement will ensure that the current features that promote atmospheric mixing in the existing plants will not be degraded by any future modifications of these plants.

Proposed Change

- Retain the requirement for insuring a mixed containment atmosphere stipulated by 50.44 (b)(2).

5.1.2.3 Requirement 50.44: Control Combustible Gases in Containment Following a Postulated LOCA

Requirement

50.44(b)(3) states the following:

> *"...(b) Each boiling or pressurized light-water nuclear reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with the capability for ... (3) Controlling combustible gas concentrations in the containment following a postulated LOCA..."*

50.44(b)(3) has several supporting requirements. Specifically, 50.44(c)(1) requires licensees to show that during the time period following the postulated LOCA but prior to effective operation of the combustible gas control system, either an uncontrolled hydrogen-oxygen recombination would not occur or the plant could safely withstand it, and if such a showing cannot be made then 50.44(c)(2) requires that plant to inert its containment. 50.44(c)(3)(ii) requires plants that relied upon a purge/repressurization system as the primary means for controlling combustible gases to install an internal recombiner or have a capability to install an external recombiner and has further requirements on the containment penetrations used with the external recombiners. 50.44(d)(1) and (d)(2) specify the quantity and rate of hydrogen assumed to be generated in the postulated LOCA. 50.44(e) requires plants whose notice of hearing on the application for a construction permit was received after 11/5/70 to have systems other than purge-repressurization systems as the primary means of combustible gas control and 50.44(f) and (g) allow plants whose notice of hearing on the application for a construction permit was received before 11/5/70 to have only purging systems if certain dose based criteria are met.

Licensees have met these requirements in a variety of ways. Many plants have classified recombiners as essential equipment and installed redundant, safety grade, internal recombiners.

Some plants do not have internal recombiners but have provisions to install an external recombiner if needed. Some plants also maintain a filtered purge system as a backup system to control post-LOCA hydrogen concentrations.

Evaluation

The risk significance of the systems used to meet the post-LOCA combustible gas requirements of 50.44 is low. There are basically three reasons for this conclusion. First, the risk of the design basis LOCA accident itself is low, as pointed out above in the discussion under 5.1.1. Second, the recombiners can only process a very limited amount of hydrogen and would be completely overwhelmed by the quantity and rate of hydrogen expected to be evolved in a more risk significant severe accident. The only useful role for recombiners is in mitigating hydrogen (and oxygen) released in the long-term from phenomena such as radiolytic decomposition of the reactor coolant and metal corrosion. Third, in some plants operation of the (backup) purge systems would be problematic in a severe accident situation as it would potentially create a direct path for fission product release outside of containment.

In addition, the provisions of this requirement are not necessary to meet any of the defense-in-depth elements (as listed in Section 2).

Proposed Change

- Eliminate 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.

5.1.2.4 Requirement 50.44: High Point Vents

Requirement

50.44(c)(3)(iii) states the following:

> *"...To provide improved operational capability to maintain adequate core cooling following an accident.......each light-water nuclear reactor shall 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...."*

Licensees have met this requirement by installing high point vents. The vent system and components are safety-grade and seismically and environmentally qualified and subject to appropriate technical specifications and maintenance, testing, and surveillance requirements

50.44(c)(3)(iii) also provides associated requirements for the high point vents:

> *"... Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of appendix A and appendix B of this part.*..."

Appendices A and B impose the safety-grade, seismic and environmental qualification requirements.

Evaluation

From the containment standpoint, the vents are actually one of the means by which hydrogen can be introduced into the containment. They were installed after the TMI accident to permit venting of non-condensible gases from the reactor coolant system that could potentially impede the operation of the emergency core cooling system. In terms of the framework for risk-informing Part 50 they are not directly related to the preventive strategy to limit the core damage frequency, since core damage precedes hydrogen generation. The vents could be instrumental for terminating a core damage accident by allowing natural circulation to occur and thus preventing further accident progression. Since continued accident progression can lead to complete core melt and vessel failure, resulting in a threat to containment integrity, the vents do have a mitigative value for reducing the likelihood for early containment failure. In any case, PRAs typically do not model scenarios in which a core damage accident is terminated as a result of using the high point vents, and therefore the risk significance of the vents is difficult to quantify. The vents do provide a means for removing non-condensible gas pockets which could impeded ECCS or natural circulation core cooling. Design requirements ensure that the potential of the vents as LOCA sources is limited.

Proposed Change

No change to the existing requirement 50.44(c)(3)(iii) and to the related regulations is proposed. Although this requirement is not directly related to mitigating the hydrogen threat to containment integrity, it has some risk-significance and moving the regulation would entail administrative costs.

5.1.2.5 Requirement 50.44: Hydrogen Control for BWR Mark I and II Containments

Requirement

10 CFR 50.44(c)(3)(i) states the following:

> *"...an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment..."*

Licensees have met this requirement by providing systems for inerting the containment. The only requirements that plants have to meet relate to the inerting system lines that penetrate containment.

Evaluation

The risk significance of inerting for the small volume containments such as Mark I and II is extremely high. Studies have demonstrated that the conditional probability of containment failure from combustible gas deflagration and detonation events would be very high if the containment was not inerted. In terms of a high-level tie to the framework for risk-informing Part 50, inerting of Mark I and II containments is related to the mitigating strategy, i.e., limit radionuclide releases and limit public health effects. The slow accumulation of oxygen generated from radiolysis could deinert the containment atmosphere in the later stages of a core melt accident, and should therefore be addressed during the severe accident management phase.

Proposed Change

- Retain an inerted containment for BWR Mark I and Mark II plants by keeping 50.44 (c)(3)(i), and address continued inerting later in the accident by providing sufficient O2 control during the severe accident management phase.

5.1.2.6 Requirement 50.44: H2 Control System for Mark III and Ice Condenser Containments

Requirement

10 CFR 50.44(c)(3)(iv) states the following:

> *"(iv) (A) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall provide its nuclear power reactor with a hydrogen control system justified by a suitable program of experiment and analysis. The hydrogen control system must be capable of handling without loss of containment structural integrity an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume)."*

Licensees have met this requirement by installing a distributed glow plug igniter system in containment. The igniters are deployed in two separate groups, each group with its own independent and separate power supplies and controls.

10 CFR 50.44(c)(3)(iv) has a number of other supporting requirements in 50.44:
- Demonstrate containment structural integrity based on actual material properties or ASME B&PV code (c)(3)(iv)(B);
- for H2 control system using post-accident inerting show containment can withstand increased pressure during the accident or following inadvertent full inerting in normal operation (c)(3)(iv)(D);
- requirements on systems and components for plants with post-accident inerting control systems (c)(3)(iv)(E);
- requirements on systems and components for plants that do not rely on inerting for H2 control (c)(3)(v)(A); and
- for plants with construction permits issued prior to 3/28/79 provide evaluation of consequences of H2 using accident scenarios acceptable to NRC that support design of control system (c)(3)(vi)(A), (c)(3)(vi)(B).

There are other regulations in Part 50 that impose associated technical requirements: references to the ASME Boiler & Pressure Vessel (B&PV) code requirements for steel containments (50.55) and written communications on accident analyses (50.4).

For both 50.44(c)(3)(iv) and the related regulations, there are associated implementation guidance documents: ASME B&PV code sections for steel containment (Section III, Subsubarticle NE-3220, Service Level C limits), and ASME B&PV Code sections for concrete containments (Section III, Subsubarticle CC-3720, Factored Load Category).

Evaluation

The risk significance of the igniter system is high in the intermediate volume Mark III and ice condenser containments. A severe accident in these plants generating significant amounts of hydrogen from the metal-water reaction (on the order of that generated in the TMI accident) would pose a severe threat to containment integrity in the absence of a hydrogen control system. However, the igniters need AC power to operate and are thus not available during station blackout

(SBO) accident sequences. Recent studies have demonstrated that the conditional containment failure probability from hydrogen combustion during SBO is very high. During an SBO sequence in an ice condenser, neither the igniters nor the air return fans are operational.

In terms of a high-level tie to the framework for risk-informing Part 50, a hydrogen control system in Mark III and ice condenser containment plants is part of the mitigation strategy, i.e., it belongs to the strategies to limit radionuclide releases and limit public health effects. At a lower level, the hydrogen control system would be used to limit the conditional large early probability and the conditional late large release probability to low values (e.g, the value 0.1 recommended in the framework for risk-informing Part 50). However, as currently configured, there is some question whether the lower level objective on the conditional probabilities can be met. The option below considers providing hydrogen control during SBO sequences, or demonstrating that these sequences are unlikely.

Hydrogen control systems have not been considered for large volume containments including large dry and sub-atmospheric containments. Risk studies for individual plants have demonstrated that the containment could withstand the pressure spike from a hydrogen combustion event and the probability of reaching a concentration where a deflagration-to-detonation transition can occur is low.

Proposed Change

- Modify 50.44 (c)(3)(iv) to include combustible gas control during all risk significant accidents (e.g., SBO sequences) in BWR Mark III and PWR ice condenser containments.

This option is aimed at ensuring that the hydrogen control system called for under the current 50.44(c)(3)(iv) would also operate during any risk significant accident (e.g., station blackout, i.e., loss of all AC power). Since all operating Mark III and ice condenser plants meet this requirement via a glow-plug igniter system, this option would translate in practical terms into requiring that all igniters (or a limited set of igniters with the number having to be demonstrated by the licensee) would be operable during risk-significant accidents (e.g., SBO). Alternatively, the implementing documents for this regulation could also allow licensees to demonstrate that the core damage frequency contribution from all sequences without hydrogen control is not risk significant, e.g., demonstrate that SBO,had a low frequency (a frequency below a threshold, such as 1E-6/year, that would be quantified in the accompanying Regulatory Guide). The rationale here is based on the discussion provided under the quantitative objectives for risk-informing regulatory requirements in the framework for risk-informing Part 50. The perspective is that for accident sequences or accident classes where one of the high level defense-in-depth strategies is precluded, more emphasis needs to be placed on the strategies that remain. Therefore for accident classes where the high-level conditional early containment failure probability goal of 0.1 specified in the framework for risk-informing Part 50 cannot be achieved a more stringent guideline needs to be imposed on their contribution to the core damage frequency.

5.2 Alternate Risk-Informed Options

The approach adopted for developing alternative risk-informed requirements is described at a high level in Chapter 2. The approach involves the following three stages/steps in the process at which risk-informed requirements can be developed:

(1) Identify the combustible gas concern

(2) Assess the defense-in-depth strategies

(3) Identify and describe the functional relationship of each strategy

Potential risk-informed options are identified and described below for each of these stages. Each proposed requirement includes consideration of the six considerations described in Section 2.2.1.

The concern is the structural failure (early or late in the accident sequence) of the containment caused by the deflagration/detonation of combustible gases. The problem that gives rise to the early containment failure concern is the rapid generation of large quantities of hydrogen generated from the reaction of zirconium clad and steam during an accident in which significant core damage occurs while it is in the reactor vessel and at the time of vessel breach . The risk significance of early containment failure is potentially very high because it can lead to the release of a large quantity of radionuclides relatively early in an accident sequence. If the release is prior to the effective evacuation of the close in population early health effects can occur. A late release is also possible because core-concrete interactions can produce large quantities of combustible gases (i.e., hydrogen and carbon monoxide) for a long time after vessel breach. If these gases ignite then late containment failure is possible and, if the sprays are not operating, a significant amount of radionuclides can be released. A late release source term to the environment is generally less severe than a corresponding early release source term, but it can still cause long-term health effects and extensive land contamination.

5.2.1 Options Dealing with Combustible Gas Concern

At this stage options can be developed, based on the characteristics noted above, to either eliminate the problem altogether or ensure that the frequency of accidents leading to significant core damage are very low.

Two options have been identified for addressing this concern in a risk-informed manner as follows:
(1) eliminate the concern
(2) lower the frequency of concern

First Option: Eliminate the Combustible Gas Concern

Proposed Change

- eliminating the concern by replacing the current regulation with a high level requirement to demonstrate that large amounts of combustible gas can not be generated at high temperature. One way of making this demonstration is via the selection of materials for the reactor core and/or the reactor coolant (e.g., the reactor cores be constructed of materials that will not react with the coolant to produce large quantities of combustible gas at high temperatures).

This requirement, however, is unlikely to be practical for the current population of operating nuclear power plants (NPPs) because it would require extensive redesign and reconstruction of the existing reactor cores. Such a massive undertaking would be difficult to justify based on cost-benefit arguments.

A risk-informed requirement of this nature however may be an option potentially applicable to the design of future reactors.

Evaluation

Implementation of such a requirement would reduce the conditional containment failure probability from combustible gases to a negligible probability. Remaining possible sources of combustible gas generation, such as steel/water interaction and core/concrete interaction, are unlikely to produce sufficient amounts of combustible gases to threaten containment integrity.

Second Option: Lower the Frequency of Concern

Proposed Change

- replace the current regulation with a high level requirement to demonstrate that the containment integrity is not challenged by accidents leading to significant core damage (and hence large quantities of combustible gas production).

Specific ways of making such a demonstration would be spelled out in a Reg Guide. The Reg Guide could specify quantitative core damage frequency targets as one way of demonstrating compliance.

Evaluation

A large number of the current population of operating nuclear power plants (NPPs) have demonstrated that the risk posed by combustion varies significantly between the various containment designs (refer to Chapter 4).

In order to meet the overall quantitative goal for large release probability (described in the framework document), by using only preventative strategies, the CDF for the individual NPPs would have to be sufficiently low (e.g., lower than 1E-5/ry as specified in the framework for risk-informing Part 50). As the risk from combustion events in plants with large dry and subatmospheric containments is relatively small (refer to Chapter 4), requiring that these plants operate with a very low CDF (e.g., below 1E-5/ry) in order to simply lower the frequency of the threat to containment from combustible gases cannot be justified based on cost-benefit arguments.

Studies have shown however that the risk from combustion is higher in ice condenser and Mark III containments, and much higher in Mark I and II containments. Therefore, before making recommendations, cost-benefit arguments should be explored in greater detail for plants with these containment designs.

5.2.2 Options based on Framework Defense-in-depth Strategies

As noted above, the framework document defines four defense-in-depth strategies for limiting accident risk at a high level, and quantifies three of them. These strategies aim to both prevent core-damage accidents (two high-level preventive strategies) and mitigate the public impact should a core-damage accident occur (two high-level mitigative strategies). Therefore, options can be developed based on each strategy. In this section each strategy is examined relative to preventing and mitigating the combustible gas concern.

Proposed Change

- Replace the current regulation with a regulation that specifies the following specific mitigative and preventative goals designed to address the combustible gas concern:
  - — demonstrate that any risk significant core damage accident does not result in an un acceptable conditional large early release probability (CLERP) and conditional large late release probability (CLLRP) as a result of combustible gases, if not
  - — demonstrate that any risk significant core damage accident does not result in an unacceptable large early release frequency (LERF) and large late release frequency (LLRF) as a result of combustible gases, if not
  - — demonstrate adequate emergency preparedness for each core damage accident class for which the above criteria are not met.

Specifics on demonstrating compliance would be provided in a Reg Guide and would be consistent with the framework guidelines. The Reg Guide could elaborate on how compliance can be demonstrated relative to the framework guidance, as follows. As the combustible gas concern is directly related to the challenge to containment integrity posed by the deflagration/detonation of combustible gases, the first strategy considered is the goal of limiting the CLERP and CLLRP to <0.1 (conditional on core damage). If this cannot be demonstrated the next option would be to use the preventative strategies to require that the LERF and LLRF goals are met. Note the early goal can be met by CLERP or LERF and the late goal can be met by CLLRP or LLRF. Consequently it is possible to meet the early and late goals with combinations of mitigative and preventative strategies i.e., LERF in combination with CLLRP. If low conditional probabilities or frequencies cannot be achieved, then the final step would be to resort to the last mitigative strategy related to emergency preparedness. These three steps are consistent with the framework approach which prefers a balance between prevention and mitigation, but recognizes that in some cases the quantitative goals of individual high level preventive or mitigative strategies cannot be met. In these cases the framework advocates more stringent quantitative goals for the remaining requirements.

The steps involved in this process can be stated as follows:

Meet mitigative strategy

- demonstrate the CLERP and CLLRP from combustible gases is sufficiently low (e.g.< 0.1as specified in the framework) for each core damage accident class

if not,
Meet a combination of the preventive and mitigative strategies

- demonstrate the LERF and LLRF from combustible gases is sufficiently low (e.g.< 1E-6) for each core damage accident class

if not,
Meet emergency preparedness criteria

- demonstrate adequate emergency preparedness for each core damage accident class for which the above criteria are not met.

Evaluation

Implementing the elements in this proposed change, would ensure that the risk from combustible gases would be low since their risk significance would be below the framework guidelines.

5.2.3 Options Based on Functional Relationship of Framework Defense-in-Depth Strategies

In this section functional relationships are identified with the aim of addressing the combustible gas concern. The focus is on limiting the conditional large release probability to sufficiently low values (e.g., <0.1) conditional on core damage and the generation and release to containment of significant quantities of combustible gases. Ways of preventing combustion or achieving controlled burning are presented in the Table 5.1 below:

Table 5.1 Combustible Gas Control Systems

Hydrogen Control System	Pros	Cons
Pre accident inerting	Will prevent combustion with high reliability	Cost is high and access to containment is restricted
Other options, e.g., passive autocatalytic recombiners (PARS), passive igniters	Will prevent combustion and/or prevent uncontrolled deflagration and detonation provided system functions	The reliability and effectiveness of the system has to be demonstrated and the impact of the additional pressure/temperature loads (caused by controlled deflagration) taken into account.
Combustible gas control systems: Glow plug igniters	Designed to burn combustible gas at low concentrations and prevent uncontrolled deflagration or detonations	The reliability and effectiveness of the system has to be demonstrated and the impact of the additional pressure/ temperature loads (caused by controlled deflagration) taken into account.

Other systems designed to control combustible gas (i.e., recombiners or vent-purge systems in operating plants) or remove heat from containment (i.e., sprays or fan cooler systems) have response times that cannot mitigate combustion of large amounts of combustible gas in a short time interval.

A number of combustible gas control systems, along with their pros and cons, are discussed in NUREG/CR-2726, “Light Water Hydrogen Control Manual” [5]. More recently an experimental program was conducted at the Surtsey facility at Sandia National Laboratories [6] to evaluate a PAR design developed by the NIS Ingenieurgesellschaft Mbh of Hanau, Germany [7].

The concern posed by combustion resulting from full-core melt accidents depends on the specifics of the containment design. Extensive research and analysis has demonstrated (refer to Chapter 3) that the six containment designs currently in use in the US can be grouped into the following three classes for the purposes of assessing the impact of combustion.

BWRS with Mark I and Mark II Containments

Risk analyses, performed for plants with Mark I and II containment buildings, model the containment atmospheres as inert. Containment failure due to combustion is therefore found to be not significant in most PRAs. However, sufficient calculations have been performed for these small volume containments, assuming that they are not inerted, to demonstrate that the containment will be severely challenged by combustion during a full core meltdown accident. Given the large concentration of combustible gases that a core damage accident in these plants could cause, the likelihood of early containment failure from combustion is very high for these containment designs. This means that an igniter system designed to burn combustible gas in a controlled manner would not prevent failure in these containment designs. Also, a post-accident inerting system is not practical because it would impose significant pressure loads caused by the addition of inert gas. Therefore, it appears that pre-accident inerting is the optimum strategy for mitigating combustion.

Although combustion would be prevented during the early stages of a core melt accident if the containment is inert, oxygen is gradually generated during accidents of this type and a combustible mixture could eventually be reached late (on the order of days) in the accident sequence. A means of controlling combustible gases during the long term severe accident management is therefore needed for these containment types even if the containment atmospheres are inert during normal plant operation.

BWRs with Mark III and PWRs with Ice Condenser Containments

For plants with Mark III and ice condenser containments, existing PRA analyses include the igniter systems in the plant model. Nevertheless, combustion was still found (refer to Chapter 4) to be a significant contributor to early containment failure in some of the analyses, mainly from station blackout sequences. It is likely that the combustion contribution to early failure would increase significantly for other sequences if the igniters were not present. Therefore, it appears that an igniter system capable of operating during SBO accidents is appropriate for these containment designs providing it can be demonstrated that controlled burning does not threaten containment integrity.

PWRs with Large Volume and Subatmospheric Containments

Numerous risk studies have demonstrated that combustion is not a significant threat during the first 24 hours of a core melt accident in large volume, dry containments. Generic Issue-121 [8] addressed the problem of combustible gas control in large, dry containments housing PWRs. The resolution of this issue was that combustion was not a failure threat for large, dry containments and that there was no basis for requiring combustible gas control measures, such as inerting or igniters, in these plants. While combustion is not a threat to containment integrity during the first 24 hours of a core meltdown accident in these containments, for severe accident management purposes it should be remembered that significant quantities of combustible gases (hydrogen and CO) could accumulate over a long period of time (on the order of days) to significant concentrations. Severe accident management strategies should account for a threat to containment integrity from a combustion event late in a core meltdown accident sequence.

Based on the above discussion the following option, based on the functional relationships of the Framework defense-in-depth strategies, is proposed that would specify specific systems to be installed or requirements to be met for the three different containment classes to either prevent or control combustion.

- *Identify Functional Relationships to address the Combustible Gas Concern*

Proposed Change

- Replace the current regulation with a regulation that specifies specific requirements to address combustible gas concern for each containment type. The following control systems or requirements would be included in the alternate option:

  (a) An inert atmosphere shall be provided for each boiling water reactor (BWR) with a Mark I or Mark II containment. Severe accident management strategies should consider that oxygen can be generated in the long term during a core meltdown, perhaps necessitating a system to prevent or control combustion late in the accident sequence.

  (b) A combustible gas control system shall be provided for each BWR with a Mark III containment and each pressurized water reactor (PWR) with an ice condenser containment. The effectiveness of the control system shall be justified by a data from a suitable program of experiment and analysis.

  (c) Licensees with a PWR with a large volume or subatmospheric containment design should include, in their severe accident management plans, strategies to demonstrate that the containment can safely accommodate a specified combustible gas source term representative of a full-core meltdown accident.

If the option includes provision for a combustible gas control system and associated systems then the attributes of the system(s) will also be described. At a minimum such a system(s) should provide, with reasonable assurance, that:

- Combustible gas concentrations uniformly distributed in the containment do not exceed 10% assuming the specified combustible gas source term, or that the post-accident atmosphere will not support combustion,
- Combustible gas concentrations will not collect in areas where unintended combustion or detonation could lead to loss of containment integrity or loss of appropriate mitigating features,
- Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety functions during and after being exposed to the environmental conditions attendant with the specified combustible gas source term including the environmental conditions created by activation of the control system,
- Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will also function for all risk important full-core meltdown accident sequences,
- If the method selected for combustible gas control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Evaluation

The above option is derived from risk insights, consistent with the framework document (i.e. it is based on the high level mitigative strategies) and supports the defense-in-depth philosophy. The option, as constructed, is very specific and specifies the systems to be installed or requirements to be met for each containment that will ensure that the deterministic large release goal of CP of < 0.1 is achieved.

5.3 Summary

In Section 5.1 the three analytical and six physical requirements of the current 10 CFR 50.44 were examined, and risk-informed options to change or replace each individual requirement were identified and discussed. The three analytical requirements were modified based on the results of risk studies which suggest that full core melt accidents dominate the combustible gas risk, and that all potential sources for combustible gas generation should be accounted for in a realistic manner when assessing the robustness of a plant relative to the combustible gas concern. The risk-informed analytical requirements are: (1) full core melt accidents must be considered, (2) combustible gas generation from metal/water reaction and core/concrete interaction must be accounted for, and (3) realistic rates and amounts of combustible gas should be assessed. Based on these considerations the six physical requirements of the current rule were risk-informed as well, and , taken together, the risk-informed options developed in Section 5.1 form a complete and comprehensive alternate means to address the threat to containment integrity posed by combustible gases

In Section 5.2 the combustible gas concern for containments was examined without reference to the existing 10 CFR 50.44, but again with the objective of developing a risk-informed alternate to the existing rule. The same risk-informed analytical requirements formulated in Section 5.1, based on current understanding of the combustible gas risk, apply here also. Four options for a risk-informed combustible gas rule were identified in Section 5.2. Two of these options were dismissed as not feasible for current reactors, while the other two were examined further. One of these latter two was based on the framework defense-in-depth strategies while the other one was based on the framework functional relationships supporting the defense-in- depth strategies. The option based on the framework defense-in-depth strategies represents a comprehensive, performance-based means to control combustible gases, as discussed further in Chapter 6. The option based on the functional relationships can be seen to lead to the same requirements as the sum of the options derived in Section 5.1, and therefore does not need to be discussed separately.

A risk-informed alternative to the existing 10 CFR 50.44 regulation is one which achieves compliance with the three risk-informed analytical requirements, which articulate the current state of knowledge regarding the combustible gas threat. Practically this would mean demonstrating that the NRC developed, realistic combustible gas source term, can be accommodated without unacceptably high risk to public safety. Such a demonstration could be accomplished by either employing the sum of the options developed in Section 5.1, or by adopting the option of Section 5.2 based on the framework defense-in-depth strategies. Both methods lead to the implementation of a risk-informed alternative to the existing 10 CFR 50.44, as discussed further in Chapter 6.

5.4 References

1. US NRC, “Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident,” Regulatory Guide 1.7, Revision 2, November 1978.

2. US NRC, "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.

3. US NRC, "Emergency Planning and Preparedness for Nuclear Power Reactors," Regulatory Guide 1.101, Revision 3, August 1992.

4. US NRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

5. Allen L. Camp, et al., "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, SAND 82-1137, Sandia National Laboratories, August 1983.

6. T.K. Blanchat and A. Malliakos, "Performance Testing of Passive Autocatalytic Recombiners," NUREG/CR-6580, SAND 97-2632, Sandia National Laboratories, 1998.

7. NIS Ingenieurgesellschaft, "NIS Control Module for Hydrogen Removal in Containment Atmosphere," Information Brochure, Hanau, Germany, 1992.

8. US NRC, "Resolution of Generic Safety Issues," NUREG-0933, 1999.

1. EVALUATION OF RISK-INFORMED ALTERNATIVE TO THE EXISTING 10 CFR 50.44

6.1 Achieving Goals of Risk-Informed Alternative

In Chapter 5 options were identified for the development of a risk-informed alternative to the existing combustible gas rule specified in 10 CFR 50.44. Two paths were followed in identifying options: one evaluated the existing set of technical requirements for either elimination, modification, or enhancement, depending on how well they addressed the concern the rule focused on; the other applied the four framework strategies to identify performance-based means to address the concern without regard to the existing requirements.

However, both paths are grounded in the same considerations. The current state of knowledge regarding the threat to containment integrity from combustible gases was reviewed, based on available risk studies and industry experience, and the three analytical requirements, which provide the foundation for a risk-informed alternative, were established. These are that any risk-informed alternative should account for: (1) full core melt accidents, (2) combustible gas generation from metal/water reaction and core/concrete interaction, and (3) realistic rates and amounts of combustible gas generated. Furthermore, all options are consistent with the quantitative guidelines of the framework and are based on proven technology.

The risk-informed alternative to the existing 10 CFR 50.44 is shown in Figure 6.1. As the figure illustrates, the objectives of the alternative, embodied in the three analytical requirements, can be met by two different methods, corresponding to the two paths discussed above.

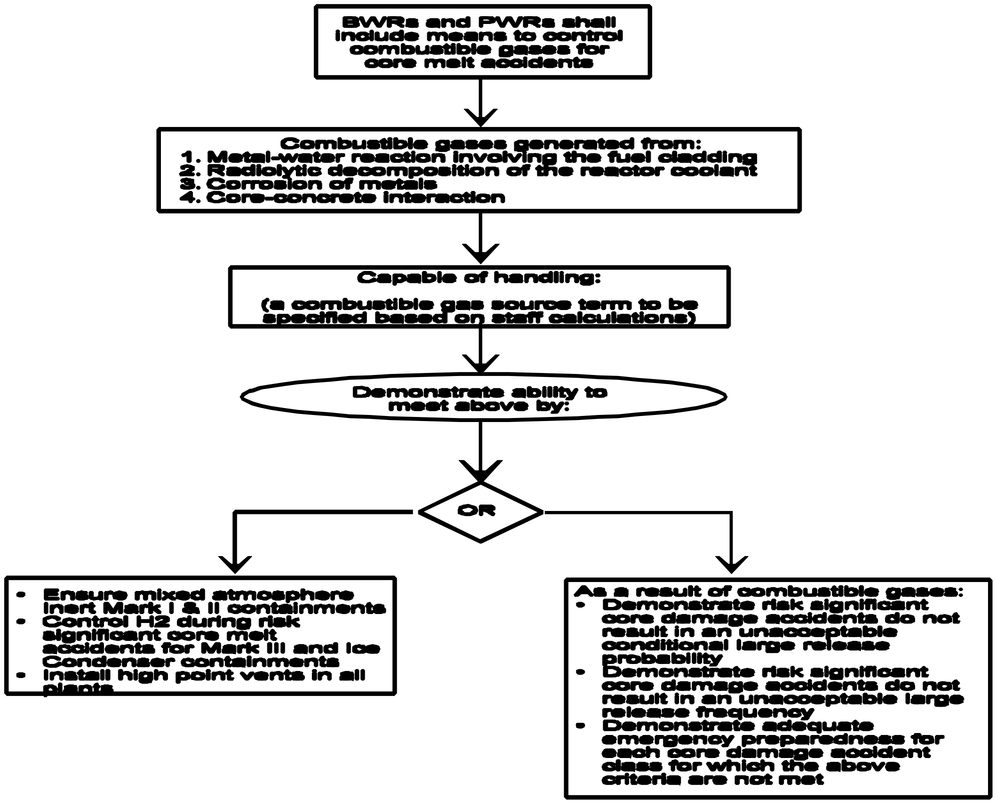

Figure 6-1 Risk-Informed Alternative for 10 CFR 50.44

First Method

The method summarized in the box on the left side is the result of eliminating, modifying or enhancing the physical requirements of the existing rule. This was done, consistent with the framework, by examining how well each existing requirement addressed the combustible gas concern. As discussed in Section 5.1, the result of this examination led to the following suggested changes to the existing physical requirements:

- Eliminate the requirement for measuring hydrogen concentration by removing 50.44 (b)(1), redundant to NUREG-0737 [1], Item II.F.1, Attachment 6, 50.47, and Part 50 Appendix E. In addition, given the need for the capability of establishing hydrogen concentration levels, under degraded core conditions, for long term accident management, recommend that related regulations be revised to remove continuous measuring and safety-grade requirements and call for an increased measurement range.

- No change to the requirement for insuring a mixed containment atmosphere, i.e., retain 50.44 (b)(2).

- Eliminate 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.

- No change to the existing requirement of 50.44(c)(3)(iii) regarding the high point vents and to the related regulations is proposed. Although this requirement is not directly related to mitigating the hydrogen threat to containment integrity, it has some risk-significance and moving the regulation would entail administrative costs.

- Retain an inerted containment for BWR Mark I and Mark II plants by keeping 50.44 (c)(3)(i), and address continued inerting later in the accident by providing sufficient O2 control during the severe accident management phase.

- Modify 50.44 (c)(3)(iv) to include combustible gas control during all risk significant accidents (e.g., SBO sequences) in BWR Mark III and PWR ice condenser containments.

Some implications of this method for NRC are summarized in Table 6.1 while some implications for industry are listed in Table 6.2. The tables present just a preliminary assessment. Implications for both the industry and the NRC will eventually have to be carefully evaluated via a Regulatory Analysis.

Table 6.1 First Method, NRC Implications

Item	yes/no	Description/Comments
Rule change	Yes	10 CFR 50.44 would be revised by making the changes indicated in Section 5.1, and summarized above, to both the analytical and physical requirement contained in the current rule.
Impact on other regulations	Yes	NUREG-0737 would be revised to allow commercial grade monitors. Part 50.47 and Appendix E may be revised.
Revise/modify implementing documents	Yes	Existing regulatory guidance on safety grade monitors in Regulatory Guide 1.97 would be revised. Regulatory guidance on recombiners will need modification.

Table 6.1 First Method, NRC Implications

Item	yes/no	Description/Comments
Create implementing documents	Yes	New regulatory guides would be needed on providing acceptable methods for compliance with different parts of the revised regulation.
Analysis	Yes	A realistic combustible gas source term, as specified in the third risk-informed analytical requirement, will have to be developed by the NRC staff.
Review	Yes	Licensee submittals will need to be reviewed to verify compliance. If the analysis is based on recommended guidance the review should be straightforward. If an alternative approach is selected, a longer review will be required.
Inspection	Maybe	Depends on way in which compliance is achieved.

Table 6.2 First Method, Licensee Implications

Item	yes/no	Description/Comments
Equipment	Maybe	Elimination of monitoring requirement would allow commercial grade monitors for Appendix E concerns. Changes would allow removal of recombiners, and purge systems. Hardware changes would be needed to make igniters operable during SBO.
Analysis	Yes	If the licensee elects to use recommended methods for demonstrating compliance then a limited amount of analysis may be needed, depending on the plant type. If the licensee elects to use an alternative method to demonstrate compliance significant additional analysis may be required.
Maintenance/inspection	Maybe	Will depend on the way compliance is achieved
Tech Specs	Maybe	Remove tech specs from monitors and recombiners and vent/purge systems.
Procedures/Training	Maybe	Will depend on the way compliance is achieved

Implicit in this alternative is the assumption that large dry and subatmosppheric containments are not challenged by combustible gases, i.e. that the conclusions of Generic Issue 121 would still apply if the capacity of these containments were assessed against a realistic combustible source term. Some confirmatory analysis may be needed.

The development of a realistic combustible gas source term by the NRC staff is discussed further in Section 6.2.

Second Method

A second method to achieve the goals of the risk-informed alternative is summarized in the box on the right hand side of Figure 6-1. This method was derived from the defense-in-depth strategies contained in the framework document for risk informing Part 50. The proposed change would be to replace the current regulation with a regulation that specifies specific mitigative and preventive goals based on the defense-in-depth strategies, that, if met, would address the combustible gas concern.

Licensees would be asked to demonstrate that:

- any risk significant core damage accident does not result in an unacceptable conditional large early release probability (CLERP) and conditional large late release probability (CLLRP) as a result of combustible gases, if not then demonstrate that
- any risk significant core damage accident does not result in an unacceptable large early release frequency (LERF) and large late release frequency (LLRF) from combustible gases, if not then demonstrate that
- adequate emergency preparedness is in place for each core damage accident class for which the above criteria are not met.

The specific means of demonstrating that the goals are met would be outlined in a Reg. Guide and would be consistent with the framework guidelines. For example the early goal can be met by CLERP or LERF and the late goal can be met by CLLRP or LLRF Consequently it is possible to meet the early and late goals with combinations of mitigative and preventative strategies, i.e., LERF in combination with CLLRP. The steps involved in this process can be stated as follows:

Meet mitigative strategy

- demonstrate the CLERP and CLLRP from combustible gases is sufficiently low (e.g.< 0.1as specified in the framework) for each core damage accident class

if not,

Meet a combination of the preventive and mitigative strategies

- demonstrate the LERF and LLRF from combustible gases is sufficiently low (e.g.< 1E-6) for each core damage accident class

if not,

Meet emergency preparedness criteria

- demonstrate adequate emergency preparedness for each core damage accident class for which the above criteria are not met.

These three steps are consistent with the framework approach which prefers a balance between prevention and mitigation, but recognizes that in some cases the quantitative goals of individual high level preventive or mitigative strategies cannot be met. In these cases the framework advocates more stringent quantitative goals for the remaining requirements.

If the above goals for large release are met, it is very likely that the plant will have risks consistent with the quantitative health objectives for the risk of early and late fatality of the Commission's Safety Goal Policy. This method embodies the high level defense-in-depth strategies articulated in the framework for risk-informing Part 50, specifically the two mitigation strategies consisting of limiting the radionuclide releases during core damage accidents and limiting public health effects due to core damage accidents. At a lower level in the hierarchy, this method is also based on the reactor safety cornerstones of ensuring the integrity of the containment boundary and the adequacy of the emergency preparedness functions.

This second method has several ways of addressing the combustible gas concern as discussed below.

Demonstrate the CLERP and CLLRP of containment failure from combustible gases is sufficiently low (e.g., < 0.1)

The framework document recommends a goal of limiting the CLERP and CLLRP to <0.1 conditional on core damage. The option is derived from risk insights, consistent with the framework document (i.e. it uses the high level mitigative strategies) and supports the defense-in-depth philosophy. However, it could be difficult to demonstrate that this goal has been met given the significant amount of uncertainty associated with predicting containment performance during core melt accidents. This option, although it appears to be flexible, would likely require significant regulatory guidance in the form of supporting documentation. For example guidance would have to be provided on the combustible gas source term (refer to Section 6.2) and on acceptable methods for demonstrating that the conditional probability goal has been met.

Acceptable methods for demonstrating that the conditional probability goal has been met could be probabilistic or deterministic in nature. Currently a PRA standard is being developed which if endorsed by the NRC could potentially form the basis for an acceptable approach for demonstrating that this goal has been met. The PRA standard currently being developed by the American Society of Mechanical Engineers (ASME)however includes only a simplified approach (which focuses on estimating Large Early Release Frequency (LERF)) for performing a level 2 PRA. The current level 2 approach in the standard is therefore only suitable for generating a simplified estimate of LERF and is not capable of demonstrating that the CLLRP goal has been met. A deterministic approach to demonstrate that the goal has been met would be to specify a combustible gas source term to containment and prescribe a method for calculating the CLERP and the CLLRP.

The combustible source term has two components, namely the total quantities of the gases to be considered and the rate of release to containment. The quantities of gases to be considered should appropriately reflect those conditions that would be expected in containment during a core melt accident. The existing regulations (refer to Chapter 3) were written to mitigate accidents like TMI-2 in which the reactor core is damaged but retained within the reactor coolant system (RCS). The combustible gas source term was therefore restricted to hydrogen (i.e., core-concrete interactions were assumed not to occur) and the amount assumed to be generated was based on a metal-water reaction limited to 75% of the clad surrounding the active fuel region. The rates of hydrogen and steam assumed to be released to containment were also determined from analyses of accidents in which core damage was terminated in-vessel.

Analyses performed (refer to Chapter 4) since TMI-2 have shown that accidents in which the core melts through the reactor pressure vessel (RPV) can pose a more severe threat to containment integrity (and thus are more risk-significant) than if the damaged core is retained within the vessel. This implies that the proposed option should address full-core meltdown accidents in which significantly more hydrogen (i.e. perhaps more than a 100% metal water reaction) and also CO may be generated. In addition, the combustible gases and steam flow rates to containment have to reflect the rapid blow-down rates associated with reactor pressure vessel (RPV) failure if it occurs at high pressure.

Demonstrate the LERP and LLRP from combustible gases is sufficiently low (e.g.,< 1E-6)

If the above criteria related to CLERP and CLLRP cannot be met then conformance can be demonstrated by calculating the LERF and LLRF and comparing the results to the above goal. Using frequencies to demonstrate conformance introduces the two preventative strategies discussed in the framework document in addition to the one mitigative strategy used for the previous two options. In addition, as the focus is on calculating frequencies for this option the level 2 approach in the PRA standard should be an appropriate way of demonstrating that the LERF goal is met provided that NRC endorses the method.

The above numerical goals, when referring to conditional probabilities of release and release frequencies from ”combustible gases,” include both direct and indirect failures and releases resulting from combustible gases. In other words, containment failures and releases which arise because combustible gas phenomena lead to failure of mitigating systems needed to achieve and maintain safe shutdown and/or containment integrity, are included.

Demonstrate adequate emergency preparedness

If the above criteria cannot be met then, by resorting to the last mitigative strategy identified in the framework document, conformance would have to be demonstrated by ensuring adequate emergency preparedness. This option would focus on those accident sequences for which the criteria could not be met. Hence it would imply that the emergency preparedness strategies used to demonstrate adequate preparedness would focus on scenarios consistent with such accident sequences. For example, emergency preparedness plans, drills, and exercises would be held under simulated conditions consistent with an SBO scenario.

Some NRC and licensee implications of the second method for achieving the risk-informed alternative goals are shown in Tables 6.3 and 6.4 respectively.

Table 6.3 Second Method, NRC Implications

Item	yes/no	Description/Comments
Rule change	Yes	This alternative would replace the existing rule with a new rule based on four preventative and mitigative strategies
Impact on other regulations	No	The new rule would not reference other existing regulations, but depending on how the quantitative guidelines of this alternative are met the licensee may have to be in compliance with related regulations (see impact on licensee, below).
Revise/modify implementing documents	No	Depending on how the quantitative guidelines of this alternative are met the licensee may have to be in compliance with related guidance (see impact on licensee, below).
Create implementing documents	Yes	New guidance would have to be provided on acceptable methods for demonstrating compliance with the conditional probability goals. New guidance would also have to be provided on an acceptable approach for demonstrating adequate emergency preparedness for those accident sequences for which the above criteria can not be met.

Table 6.3 Second Method, NRC Implications

Item	yes/no	Description/Comments
Analysis	Yes	Analysis would be needed to develop an appropriate source term.
Review	Yes	If the methods for demonstrating compliance and the recommended source term contained in the implementing documents are used then the review process should be relatively straight forward. If however an alternative approach is used to demonstrate compliance with the criteria then a significant review process would be needed.
Inspection	Yes	The way in which compliance is achieved (characteristics of the system installed) will determine the level of (i.e., added or reduced) inspection needed.

Table 6.4 Second Method, Licensee Implications

<table>
<tr><th>Item</th><th>yes/no</th><th>Description/Comments</th></tr>
<tr><td>Equipment</td><td>Yes</td><td>The way in which compliance is achieved will determine what, if any, equipment is needed. If needed, this would be a backfit issue where justification would likely require cost/benefit analysis.</td></tr>
<tr><td>Analysis</td><td>Yes</td><td>If the licencee uses the recommended methods for demonstrating compliance and the specified source term contained in the implementing documents then minimal analysis will be needed

If however the licencee elects to use an alternative approach to demonstrate compliance with the criteria then significant additional analysis could be needed.</td></tr>
<tr><td>Maintenance/inspection</td><td rowspan="3">Yes</td><td rowspan="3">The way in which compliance is achieved (characteristics of the system installed) will determine the level of additional maintenance/inspection, the role of technical specifications, and need for procedures/training.</td></tr>
<tr><td>Tech Specs</td></tr>
<tr><td>Procedures/Training</td></tr>
</table>

The collective characteristics of both methods reflect the common insights gathered from plant specific PRAs, and industry experience regarding risk significance and unnecessary burden. Besides PRA insights and industry experience, the proposed alternative also exhibits the characteristics noted in Chapter 2:

- consistency with the quantitative guidelines identified in the framework document
- reasonable cost burden
- proven technology
- suitability for performance-based monitoring

The second method discussed above can be related directly to the quantitative guidelines of the framework. However, both methods clearly emphasize measures to reduce conditional probability of early and late large releases, in keeping with the framework goals.

The first method (left hand box in Figure 6.1) represents a more prescriptive approach to combustible gas control than does the second method. The first method requires either specific combustible gas control systems, or specifies fairly detailed requirements for such systems. The second method poses the combustible gas control requirements in terms of goals to be met, rather than in terms of specific measures to be implemented. In that sense, the second method is more performance based, since it relies on a measurable outcome to be achieved and provides flexibility for a licensee as to how to achieve this outcome. The second method, as described here, meets the four point test for performance based regulation: (1) it is based on measurable or calculable parameters, i.e., on conditional probabilities or on frequencies, (2) objective criteria to assess performance are established based on risk insights, deterministic analyses and/or performance history, i.e., criteria were established based on PRA insights, the TMI-2 accident, and a number of experimental programs, (3) licensees have flexibility to determine how to meet the established performance criteria in ways that will encourage and reward improved outcomes, i.e., licensees are free to use any number of means at their disposal to meet the combustible gas threat to containment integrity as long as the specified probability and/or frequency goals are met, (4) a framework exists in which the failure to meet a performance criterion, while undesirable, will not in and of itself constitute or result in an immediate safety concern, i.e., straying above the specified probability and/or frequency goals will not be an immediate cause for alarm if the situation is addressed and corrected. It should be noted, however, that method one can also be made performance based to varying degree, depending on the guidance in the implementing documents accompanying the requirements of that method.

6.2 Required Supporting Analyses

Two methods for risk-informing the existing Part 50.44 requirements are described in Section 6.1. The first method is based on modifying, eliminating or enhancing the existing requirements in the rule. The second method does not review the existing requirements, instead it derives risk-informed requirements based on the defense-in-depth strategies contained in the framework document. It is anticipated that if this risk-informed alternative is adopted that operating plants will select the first method whereas the second method may be more attractive to future reactors. However, both methods need the development of a realistic combustible gas source term. The context in which the source term will be used and the attributes of the source term are discussed in this section.

First Method

The requirements proposed in the first method were based on the three analytical requirements (i.e., a realistic combustible gas source term) described above in Section 6.1. The staff will perform supporting analyses to define a realistic combustible gas source term, which will be used by the staff to confirm the appropriateness of the proposed requirements. The calculations will use the best available calculational methods for severe accidents that include in-vessel (and ex-vessel) hydrogen and CO generation. It is anticipated that the analyses will not change any of the proposed requirements or impose new requirements on plants with Mark I or Mark II containments (provided the containment atmospheres continue to be inert during operation) or on plants with large dry or sub-atmospheric containments.

However, for plants with ice condenser or Mark III containments, if the first method is adopted then the licensees will have to provide combustible gas control during all risk significant accidents. Existing igniter systems provide effective combustible gas control but are not available during SBO sequences. SBO sequences can be addressed in the following manners:

Demonstrate SBO sequences are not risk significant (e.g., CDF<1E-5/year)
if not,
provide additional back-up power to the igniter system so that they function during an SBO.

If not,
demonstrate that containment integrity is not challenged assuming a realistic combustible gas source term.

The licensee can develop a realistic combustible gas source term or use the source term developed by the NRC.

Second Method

As noted above it is anticipated that the second method of risk-informing 10 CFR 50.44 will be more useful for future reactors. Under these circumstances it is expected that realistic combustible gas source terms will be developed by the applicant. The NRC developed source term is therefore not expected to be used and in fact may not be applicable (depending on the proposed reactor design).

However, licensees of operating plants may also use this second method for risk-informing 10 CFR 50.44. Under these circumstances the licensee could use the NRC source term or develop their own plant-specific source term.

6.3 References

1. USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

Attachment 3

Comparison to R. Christie's Petition for Rulemaking

On November 9, 1999, Mr. Robert Christie of Performance Technology submitted a "request for proposed rulemaking" to the staff on 10 CFR 50.44 ("Standards for Combustible Gas Control System in Light-water-cooled Power Reactors"). As discussed in a January 4, 2000, letter from S. Collins to Mr. Christie, his request has been considered as part of the staff's study of possible risk-informed changes to 10 CFR 50.44. The staff's recommended risk-informed alternative to 10 CFR 50.44 addresses Mr. Christie's request. A comparison of Mr. Christie's request with the staff's recommendation follows:

1. Mr. Christie proposes that the hydrogen source term be based on realistic calculations for accidents with a high probability of causing severe reactor core damage. The staff recommendation is for the hydrogen source term to be based on the more likely severe accident challenges as defined by the framework (i.e., sequences with CDF greater than 10^{-5}/ry).

2. Mr. Christie requests elimination of the requirement to monitor hydrogen concentration; the staff also recommends elimination of this requirement.

3. Mr. Christie did not address the requirement of insuring a mixed atmosphere; the staff recommends retaining this requirement.

4. Mr. Christie's petition and the staff's recommendation both include elimination of the requirement to control combustible gas concentration resulting from a postulated-LOCA.

5. Mr. Christie's petition and the staff's recommendation both include retaining the requirement to inert Mark I and II containments.

6. Mr. Christie's petition and the staff's recommendation both include retaining the requirement for high point vents.

7. Mr. Christie proposes, for licensees with Mark III and ice condenser containments, that the hydrogen control system be capable of meeting a specified performance level. This request does not address the potential vulnerability during station blackout conditions in which the hydrogen control system would not be available (i.e., the igniters are ac dependent). The staff recommends that licensees control hydrogen during risk-significant core-melt accidents in such a way that if station blackout is risk significant, hydrogen combustion would be controlled.

8. Mr. Christie proposes that facilities with other types of containments *"must demonstrate that the reactor containment (based on realistic calculations) can withstand, without any hydrogen control system, a hydrogen burn for accidents with a high probability of causing severe core damage."* The staff believes its recommendation of using risk information and plant-specific analysis to demonstrate containment performance is equivalent to Mr. Christie's proposal.

9. Mr. Christie did not address the concern of combustible gases in the long-term. The staff's recommendation is that long-term control be included as part of the licensee's SAMGs.