D910517

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED CRITERIA TO ACCOMMODATE SEVERE ACCIDENTS IN CONTAINMENT DESIGN

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we discussed development of criteria that would incorporate explicit consideration of severe accidents into requirements for containment design. This matter was also considered during our meetings in December 1990 and in January, February, March, and April 1991. The Committee had also discussed this matter in a number of previous meetings, including discussions with the Commission, the latest on November 8, 1990. In addition, we have had the benefit of discussions with a large number of experts on containment and severe accidents, including representatives from industry, private consultants, the NRC staff, and national laboratories, in a series of ACRS joint Containment Systems and Structural Engineering Subcommittee meetings over the past three years. The Commission had earlier requested an ACRS study of this matter (see Staff Requirements Memorandum of July 28, 1988) based on discussions during an ACRS meeting with the Commission on July 14, 1988.

Our purpose in writing this report is to describe and recommend a possible course by which the NRC could develop an improved set of requirements for the design of containment systems for future nuclear power plants. These requirements would include definition of specific challenges posed by severe accidents. They would be promulgated by revisions and additions to 10 CFR Part 50, primarily to Appendix A, "General Design Criteria for Nuclear Power Plants." Implementation also would require new regulatory guides (RGs). More detail about rule changes and regulatory guides is provided in the Appendix.

We intend this to be a description of a general approach that could be taken. Guidelines for the regulatory guides are provided primarily to illustrate that approach. Final detail and quantification should be developed and justified by the staff with input and review by industry and the reactor safety community.

The new requirements would be applicable to future plants, those not yet designed. We would exclude the "evolutionary" LWRs, for which designs are well advanced. We believe the new criteria can and should be adopted for use in the development and licensing of the "passive" plant designs.

An alternative or interim approach would be to adopt the general process we propose as an extension of the "Policy Statement on

Severe Accidents Regarding Future Designs and Existing Plants" published in August 1985. This could be more easily and rapidly adopted, in comparison with the rulemaking approach, as a guide for designers and staff reviewers, and as a basis for design certification. A disadvantage is that the "policy" approach would be subject to less rigorous reviews and more limited input from the general body of available expertise on severe accidents and containment performance. We recommend the rulemaking approach.

Future licensing responsibilities of the NRC may include nuclear power plants other than LWRs. Our proposal is for application only to LWRs. As discussed above, we propose that new containment requirements be implemented through changes in appropriate sections of the General Design Criteria. The introduction to 10 CFR Part 50, Appendix A (issued in 1971), states that these criteria apply for "water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." There are some general principles that could be applicable to other types of plants. Such application is, however, a task for another day.

There will be debate over our proposal. It will center on the question of whether it is better to continue with the present set of requirements, which it might be argued are good enough, or to develop requirements that reflect what has been learned about severe accidents over the past decade. A classical conflict between short-term and long-term costs and benefits exists. We recommend that development of new containment design criteria proceed along the lines we have proposed. We believe that benefits in safer and more efficiently designed plants and in stabilization of an important part of the regulatory process will be substantial. We look forward to the opportunity to interact with you and the staff on this important subject.

Sincerely,

David A. Ward
Chairman

APPENDIX

Proposed Criteria to Accommodate Severe Accidents in Containment Design

Table of Contents

I. BACKGROUND

The primary purpose of the containment and its associated systems in an LWR plant is to mitigate the consequences of severe accidents, those which involve fuel melting and an abundant release of fission products. Other important purposes of the containment include: housing the nuclear steam supply system and protecting it from external threats, shielding the environment from radiation emanating from the reactor system, and mitigating the releases of radioactive substances caused by normal operation or incidents of lesser scope than severe accidents.

Although this primary purpose has been recognized from the beginning, and is perhaps obvious, existing NRC requirements do not account for many severe accident phenomena that could challenge a containment's ability to perform its function.

In the early 1960s, licensing authorities and the reactor safety community (including the ACRS) recognized that the risk of a severe accident was real, but remote and largely undefined. Rather than await the results of what was seen to be a long and difficult research effort to understand more about severe accidents, a decision was made to use a surrogate accident as a design basis for the containment and to move forward with the development of nuclear power. That surrogate accident, a sudden large-break LOCA, coupled with the siting criteria in 10 CFR Part 100, has been the basis for LWR containment design ever since.

During the 1979 accident at Three Mile Island 2, a containment designed to the surrogate requirements functioned effectively to protect the public. On the other hand, severe accident research and risk assessments performed since 1979 indicate that a broad range of high-energy loads and fission product releases, more severe than at Three Mile Island 2, might threaten containment systems. There are indications that certain unlikely severe accident challenges could cause containments to fail, and lead to the release of health-threatening quantities of fission products. While the predicted risk from those accidents is small, uncertainty in quantification of the risk is large. Improvements in the design of containments could reduce both the risk and the uncertainty.

II. REVIEW OF EXISTING CONTAINMENT REQUIREMENTS

Formal criteria by which acceptable reactor containments were to be designed and built were established by the Atomic Energy Commission in the 1960s and 1970s. General Design Criteria for water-cooled nuclear power plants (10 CFR Part 50, Appendix A), promulgated in 1971, included the following requirements relating to containment:

Criterion 16 specifies "an essentially leak-tight barrier" between the reactor systems and the environment as one of "multiple fission product barriers."

Criteria 38 through 40 require systems to remove decay heat from the containment to negate pressure buildup that would otherwise result.

Criteria 41 through 43 provide for a system to remove fission products from the containment atmosphere to reduce the consequences of ongoing leakage.

Criterion 50 requires that the containment structure be able to accommodate "the calculated pressure and temperature conditions resulting from any loss-of-coolant accident." This is to be accomplished "without exceeding a design leakage rate and with sufficient margin." It states that the margin should reflect consideration of (1) potential energy sources such as energy in steam generators, limited metal-water reaction that might result from degradation but not failure of the ECCS, (2) limited information on accident phenomena, and (3) conservatism in the calculations. There is no requirement in GDC 50 to accommodate severe accidents. However, this was remedied in part by 10 CFR 50.34(f) for near term operating licenses, 10 CFR 52.47 for standard design certification, and 10 CFR 50.44 for combustible gas control.

Criteria 51 through 57 provide requirements for containment materials, testing, penetrations and isolation.

Reactor siting criteria in 10 CFR Part 100, established in 1962, indirectly determine the maximum leakage rate for which the containment is to be designed. Section 100.11 establishes dose limits for the whole body and for the thyroid. A referenced document, TID-14844, suggests amounts of radioactive material within containment that are to be assumed in calculating hypothetical doses from post-accident containment leakage. TID-14844 also suggests a leakage rate of 0.1 percent of the containment volume per day.

Additional guidance is provided in two regulatory guides originally issued in 1970. Regulatory Guide 1.3 is for BWRs and Regulatory Guide 1.4 is for PWRs. Each specifies the proportions of the elemental, particulate, and organic forms of the radioiodines that are to be assumed in making dispersion and dose calculations. These are, respectively, 91 percent, 5 percent, and 4 percent. In addition, Regulatory Guide 1.4 permits the assumption that the leakage rate from containment for PWRs is reduced to one-half the value given in technical specifications after the first 24 hours.

III. WHY NEW CRITERIA ARE NEEDED

A first purpose of new containment requirements will be to reduce the risk and uncertainty by more directly accounting for severe accident threats than is done with present requirements. This should be feasible because in 1991 more is known about the nature of severe accident threats than was known in 1971. Our proposal is simply a way of applying this improved knowledge to provide improved containment systems.

A second purpose is to clarify what is expected of applicants and to bring greater coherence to the design review and certification processes. Many severe accident considerations are now being factored into staff reviews of advanced reactor designs, but, the process by which this is done is not well defined.

A third purpose is to help ensure that containments will have greater "robustness." A containment cleverly and narrowly designed to mitigate a set of accidents that has been precisely identified may not be able to cope with the unexpected. A truly "robust" containment would have improved capability to deal with the unexpected. A containment that has been designed with explicit consideration of a more extensive set of challenges is likely to be more robust than one designed with consideration of only a limited set.

IV. PROPOSED APPROACH TO DEVELOPMENT OF NEW CONTAINMENT DESIGN CRITERIA

We have previously recommended (ACRS report of May 13, 1987 regarding Safety Goal Policy) a conditional 10 percent failure probability for the containment, reflecting our judgment about the need for assurance of containment performance. It is worth recalling that our recommendation was meant as a hedge against uncertainty, to preserve the concept of defense in depth -- itself a hedge against uncertainty. If all calculations were accurate and credible, all that would matter would be that the population of plants meets the Commission's safety goals, and the identification of containment performance as a separate item would be inappropriate. It is because quantitative risk estimates are not perfect that defense in depth is a useful philosophy, and that separate containment performance guidelines make sense.

The containment performance objective should serve as guidance to the NRC staff in judging whether requirements for containment design properly reflect the intent of the Commission as expressed in the Safety Goal Policy. The conditional containment failure probability should not be simply passed on to applicants for plant licenses. Instead, we propose a two-step process to establish new requirements.

First, the General Design Criteria (GDC) in 10 CFR Part 50 would be revised to acknowledge that containments should be designed for a range of challenges that can threaten their function during severe accidents. Several different challenges or containment loads would be defined, as discussed in Section V of this Appendix. For each, the nature of the challenge would be described in general terms; specifics and quantification would be relegated to a regulatory guide. Also, for each, a success criterion would be specified. In most cases, success would be defined simply as maintenance of the containment function for an appropriate period following the particular challenge. In addition to the GDC changes, certain other regulations concerned with containment would be modified. A summary of proposed regulatory changes is given in Table I.

Second, new regulatory guides would be developed to detail acceptable means to implement the design requirements. For the severe accident requirements of GDC 50, regulatory guides would address each challenge.

The regulatory guides would provide technical definitions of acceptable means of meeting the general design criteria for containment. What we have in mind is a relationship between each GDC requirement and its companion regulatory guide similar to the existing relationship between GDC 35, "Emergency Core Cooling," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models." GDC 35 states that a system shall provide "abundant emergency core cooling." Appendix K gives, in reasonably unambiguous language, a technical definition of the leak that must be accommodated and a definition of the terms "abundant" and "cooling."

Our revised GDC 50 would state the requirement that containments must have the capability to accommodate a specific list of challenges without loss of containment function. For each challenge, a regulatory guide would define in unambiguous technical terms, first the challenge, and second, what is meant by the term "accommodate."

The technical content of each regulatory guide should provide as complete and unambiguous a basis for containment system design as can be practically developed. For example, the criterion for capacity to accommodate hydrogen combustion might state the total amount of hydrogen to be considered, as a percentage of that which could be generated by complete oxidization of cladding in contact with active fuel, and then require a specific analysis for mixing and stratification. The regulatory guide would describe acceptable mixing models, based on containment type.

An important aspect of what we are proposing is that the NRC will take responsibility for the important technical judgments necessary to transform knowledge from severe accident research and risk assessments into criteria and requirements that can be used by a designer. This would not be done in isolation; review and input from the industry and the reactor safety community should be sought as the rule changes and regulatory guides are developed.

In the following sections, we propose revisions to the regulations relating to containment design requirements and also provide information on the content of proposed regulatory guides. Although we have not attempted to couch the GDC proposals in regulatory language, we believe that the scope of our description is close to the appropriate scope for the rule. In contrast, our proposals for regulatory guides are intended to be only the bare bones of what the guides should contain. It will be up to the staff to develop quantifications and to provide appropriate justification. We will want to interact with the staff as final details are developed.

V. RECOMMENDED CHANGES AND ADDITIONS TO 10 CFR PART 50, APPENDIX A, "GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS"

We recommend that the following General Design Criteria be changed as indicated:

Criterion 16 -- Containment Design

This criterion specifies an "essentially leak-tight barrier ... for as long as postulated accident conditions require." No changes in wording are necessary but implications of the words would be different. A regulatory guide would specify a definition for leak-tight that is consistent with the overall package of containment requirements. Existing regulatory guides suggest a leakage rate of 0.1 percent of containment volume per day. Present information about severe accidents and the role of containment suggest leakage of 1 percent may be more appropriate. In addition, the accident conditions for which such a leakage limit would apply should reflect other requirements, in particular those in the new GDC 50.

Criteria 38-40 -- Containment Cooling

These requirements would be changed to reflect the demands placed upon containment cooling systems by other new requirements, especially the proposed new GDC 50(f) and 50(g) below.

Criteria 41-43 -- Containment Atmosphere Cleanup

These requirements would be changed to reflect the demands placed upon containment atmosphere cleanup systems by other new requirements, especially the proposed new GDC 50(f) below.

Criterion 50 -- Containment Design Basis

This criterion would be extensively expanded to require that containment systems be designed to accommodate a variety of challenges that could be created by severe accident conditions. We believe that the challenges can be adequately represented by the eight examples discussed below. Each would be defined in a section [(a) through (h)], with a success criterion identified and with appropriate supporting regulatory guides. These are not meant to be accident scenarios, but are representative phenomenological challenges.

50(a) Loss of Coolant Accident (LOCA)

The containment system would have the capacity to accommodate pressure and temperature conditions resulting from the blowdown of fluid from a large break LOCA; and in the case of PWRs, from a nonconcurrent blowdown of the secondary system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the accident.

50(b) Fuel-Coolant Interaction

The containment would have capacity to accommodate missiles that could be produced by credible steam explosions within the vessel and to accommodate pressure pulses that could be produced by credible steam explosions outside the reactor vessel and within containment. Steam explosions are characterized by the rapid transfer of thermal energy from molten material to water. Where appropriate, the addition of chemical energy to the thermal energy source would be included in performance calculations.

Leakage should not exceed the rate specified in Criterion 16 following the missile impact or the pressure pulse, crediting dynamic response of the containment structure.

50(c) Hydrogen Combustion and Detonation

The containment would have capacity to accommodate pressure pulses produced by static or shock loadings resulting from the combustion or detonation of hydrogen produced during severe accidents. Hydrogen sources to be considered are the in-vessel and ex-vessel oxidation of core materials, including (1) core degradation from overheating and melting, (2) steam explosions or high pressure melt ejection in the presence of water, and (3) interaction between molten core material and concrete.

Leakage should not exceed the rate specified in Criterion

16 following the pressure pulse, crediting dynamic response of the containment structure.

50(d) Melt Attack on Containment Structure or Pressure Boundary

The containment design would preclude potential for damage to the containment pressure boundary or essential structure by direct contact of molten core material.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the melt attack.

50(e) High Pressure Melt Ejection

The containment system would have the capacity to accommodate rapid increases in static pressure and temperature caused by heating of the containment atmosphere through the direct transfer of thermal and chemical energy from molten core material ejected at high pressure into the containment, unless such ejection is precluded by design of the reactor system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the melt ejection.

50(f) Corium-Concrete Interaction

The containment system would have the capacity to accommodate the following challenges resulting from the thermal decomposition of concrete by molten corium: (1) the degradation of containment cooling and of cleanup capability due to aerosol formation, (2) slow overpressurization resulting from the evolution of noncondensable gases, (3) functional degradation of structural concrete by erosion, including basemat penetration, and (4) combustion of carbon monoxide.

Challenges to the containment should not be sufficient to render inoperable that equipment required for containment cooling or atmospheric cleanup, nor to cause leakage in excess of the rate specified in Criterion 16 or to allow any release through the basemat within an appropriate time of the onset of the corium-concrete interaction sufficient to cause significant contamination of the groundwater.

50(g) Pressurization from Decay Heat

The containment system would have the capability to accommodate the long-term buildup of pressure resulting from decay heat. This could include an appropriate containment venting system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the accident.

50(h) Elevated Temperatures

Containment penetrations, equipment necessary for accident management, essential instrumentation, and key structural components would have the capacity to accommodate exposure to elevated containment temperatures.

Exposure of the noted systems and components following exposure to elevated temperatures should not be sufficient to cause leakage in excess of the rate specified in Criterion 16 or damage sufficient to render inoperable that equipment necessary for accident management for an appropriate period following the exposure.

Criteria 51-53

No changes in these criteria are proposed.

Criteria 54-57

These would be revised to be consistent with new Criterion 58. Simplification of Criteria 54-57 may be possible.

In addition to the revisions to existing criteria, described above, we recommend the following new criteria as additions to Appendix A:

Criterion 58 -- Provision For On-Line Monitoring of Containment Isolation Status

This new criterion would be intended to reduce the likelihood of loss of containment function by continuous on-line monitoring. It must be consistent with Criterion 16.

Criterion 59 -- Role of Containment Structure in Protecting Nuclear Components Against External Threats

This new criterion would be intended to protect the nuclear steam supply system and other essential components against credible aircraft crashes, explosions, and other nonnatural threats external to the plant. Alternatively, the existing Criterion 2, which calls for resistance to extreme natural conditions, could be revised to include such threats.

Criterion 59-A - Assurance of Containment Integrity During Shutdown

This new criterion would require that containments will be designed to provide for ease of emergency closure during shutdown operation including station blackout conditions.

VI. RECOMMENDED CHANGES TO OTHER REGULATIONS RELATING TO CONTAINMENT DESIGN

10 CFR Part 100, Reactor Site Criteria

The NRC staff has in progress a study (see SECY-90-341) which would uncouple siting requirements from specifics of plant and containment design. In our report of June 13, 1990, we commented on this program and endorsed the general approach envisioned by the staff.

10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

If the allowable leakage rate for accident conditions is increased, and if on-line monitoring capability is provided and used, the requirements of Appendix J would have to be modified extensively. Significant simplification of testing requirements should be possible.

10 CFR 50.34(f) Additional TMI-Related Requirements

Additional requirements pertaining to containment design were promulgated following the TMI-2 accident and are given in 10 CFR 50.34(f). For example, a minimum containment design pressure of 45 psig is specified in one of these. These requirements also apply to standard plant designs to be considered under 10 CFR Part 52. Some of these requirements would be superseded by the expanded GDC 50.

10 CFR 50.44, Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors

Requirements in this section, intended for control of combustible gas generated during severe accidents, should be superseded by new GDC 50(c).

VII. RECOMMENDED CONTENT OF THE NEW REGULATORY GUIDES FOR GENERAL DESIGN CRITERION 50

Implementation of our recommended GDC 50 will require development of new regulatory guides. In what follows are some examples of what these guides should contain. In the final regulatory guides, to be developed by the staff, each should give acceptable values of the important parameters as well as acceptable methods for their calculation. Realistic methods of calculation should be employed. Our recommendations are keyed to the proposed GDC 50 [(a)-(h)].

50(a) Loss-of-Coolant Accident (LOCA)

This regulatory guide should address the current practices for considering LOCAs and indicate the following additions or changes. The best estimate methodology of 10 CFR 50.46 should be used. Active cooling systems should not be credited in calculating maximum containment pressures during blowdown. The effect of thermal stratification on thermal stresses in steel liners and penetrations should be considered.

50(b) Fuel-Coolant Interaction (FCI)

In-Vessel FCI

This regulatory guide should treat PWRs and BWRs separately to account for differences in core degradation processes and differing amounts of zircaloy relative to other materials in the core. For PWRs with safety depressurization systems or with low-pressure sequences of importance, in-vessel FCI should be considered. What constitutes an acceptable mechanistic treatment to establish the quantities of molten core and its temperature should be delineated. Examples of acceptable methods for calculating the mechanical energy produced by FCI should be given. Still further it should specify, for example, that missile velocity be calculated with consideration given the vent path through the downcomer and possible lower head failure. For present-day BWRs, in-vessel FCI is not expected. Future BWRs should be reviewed to be certain this conclusion is still valid.

Ex-Vessel FCI

For FCI outside the vessel, e.g., in a water-filled cavity under the reactor, somewhat different assumptions would be appropriate. Conditions at the time of vessel failure should be used to prescribe the amount and composition of the core material, and its temperature, that need to be considered for evaluation of ex-vessel FCI potential. This regulatory guide should indicate what is acceptable as well as what is an acceptable method for its calculation.

Containment designs that do not preclude water from being in the reactor cavity at the time of vessel failure must consider ex-vessel FCI. This regulatory guide should indicate acceptable methods for calculation of the amount, composition and temperature of the molten core materials at the time of failure, and the type of vessel failure and mass flow rate of molten materials. These calculated values should be used in calculating the mechanical energy produced by the FCI. The mechanical energy calculation is to be based on the same method as described above.

50(c) Hydrogen Combustion and Detonation

There will be different amounts of hydrogen generated by the different reactor types. This regulatory guide should specify the amount of metals oxidized in-vessel and ex-vessel as percentages of what is available, as well as give guidelines as to what constitutes an acceptable mechanistic method for calculating the rate and amount of hydrogen produced. Hydrogen is produced following vessel failure during (1) interactions with water in the cavity, if it exists, and (2) subsequent corium-concrete interaction. This regulatory guide should give guidance as to how much metal is oxidized in each of these two phases of the accident and give guidance to those who wish to calculate it themselves.

Hydrogen in the containment atmosphere can lead to combustion, deflagration, or detonation. All must be considered. To deal with detonation, this regulatory guide should indicate what hydrogen control methods are acceptable and give both acceptable peak pressure and pressure pulse shape with guidance as to how they can be calculated. This guide should also give examples of acceptable analysis methods for calculation of hydrogen distribution within the containment. Pressure calculations should include the effect of hydrogen burns as well as carbon monoxide from corium-concrete interaction with account taken of the timing of the various gas generation processes. The noncondensables from the corium-concrete interaction should also be considered in pressure calculations.

50(d) Melt Attack on Containment Structure or Pressure Boundary

This regulatory guide should contain acceptable values of the molten core material composition, temperature, and rate at which it pours out of the vessel breach, as well as guidelines for an acceptable analysis. Presence of water in the cavity under the reactor should be assumed if the plant is so configured. If justified by a credible spreading analysis, uniform spreading may be assumed. Otherwise, consequences of nonuniform melt depths should be considered. Appropriate heat transfer calculations should be required to establish the thermal insult to the pressure boundary or essential structures.

50(e) High Pressure Melt Ejection

This regulatory guide would apply only to PWRs and only if a depressurization system is not available. It should give guidance on what constitutes acceptable analysis for calculation of thermal energy and corium composition shown to be credible at the time of failure of the reactor pressure vessel. The regulatory guide should indicate that the amount, composition, flowrate, and temperature of the molten material be calculated by an acceptable method. The containment atmosphere should be assumed to be saturated with water vapor. Presence of water in the cavity under the reactor should be included in the analysis if the plant is so configured. Allowable amounts of de-entrainment along the flow path should be specified or methods for their calculation should be given. Oxidation of and heat transfer from the entrained debris should be based on mechanistic modeling.

50(f) Corium-Concrete Interaction

This regulatory guide would be the same for all reactor types. It should specify that a mechanistic evaluation of corium-concrete interaction be performed. The results of an acceptable core melt and vessel failure analysis, defined in this guide, should be used to define the core melt characteristics as it arrives on the reactor cavity floor. Water in the reactor cavity should be accounted for in calculations. The basemat must be shown to be thick enough to provide an appropriate interdiction time before penetration. With consideration given to timing, the contribution of combustibles and noncondensables to containment atmosphere pressure and temperature should be accounted for. Selection of concrete types that reduce gas generation and the use of refractory materials should be encouraged. Core debris control devices and filtered venting for long term pressure control should not be precluded by this guidance.

50(g) Pressurization by Steam From Decay Heat

This regulatory guide should allow for credit to be taken for the decrease in decay heat with time, for heat transfer across the containment boundary, and for heat removal by operable containment equipment. Restoration of emergency cooling should be credited after an appropriate time following the accident.

50(h) Elevated Temperature

This regulatory guide should specify that a mechanistic calculation of the containment atmosphere thermal history be made with appropriate treatment of stratification including consideration of the following: (1) hydrogen combustion, (2) high pressure melt ejection, (3) LOCAs, and (4) molten corium-concrete interaction. A detailed heat transfer analysis should be required to ensure that seals, penetrations, equipment, and other items of safety significance are not damaged. For containments with steel liners, thermal stresses induced by stratification should be considered.

TABLE I

SUMMARY OF PROPOSED CHANGES TO REGULATIONS TO INCORPORATE SEVERE ACCIDENT CONSIDERATIONS INTO CONTAINMENT DESIGN REQUIREMENTS

	Regulation	Subpart	Description of Change	Ancillary Requirements
1	10CFR50 Appendix A General Design	GDC 16 Containment Design	No change required	Regulatory Guide changed to specify maximum acceptable (e.g., 1%/day)

	Criteria		leakage rate for range of severe accident challenges	
2	10CFR50 Appendix A General Design Criteria	GDC 38 Containment heat removal	Requirement for heat removal capability consistent with require- ments of GDC 50(f) and 50(g)	None
3	10CFR50 Appendix A General Design Criteria	GDC 39 Inspection of containment heat removal system	No change required	None
4	10CFR50 Appendix A General Design Criteria	GDC 40 Testing of containment heat removal system	No change required	None
5	10CFR50 Appendix A General Design Criteria	GDC 41 Containment atmosphere cleanup	Requirement for cleanup system to function consistent with requirements of GDC 50(f)	New RG may be needed
6	10CFR50 Appendix A General Design Criteria	GDC 42 Inspection of containment atmosphere cleanup system	No change required	None

	Regulation	Subpart	Description of Change	Ancillary Requirements
7	10CFR50 Appendix A General Design Criteria	GDC 43 Testing of containment atmospheric cleanup system	No change required	None
8	10CFR50 Appendix A General Design Criteria	GDC 50 Containment design basis	Existing GDC 50 will be expanded by new 50(a)- 50(h), described below	As shown below
9	10CFR50 Appendix A	GDC 50(a) Containment	Specifies requirement that containment accommodate	New RG needed

	General Design Criteria	design basis-LOCA	LOCA or steam line failure without excessive leakage	
10	10CFR50 Appendix A General Design Criteria	GDC 50(b) Containment design basis-fuel-coolant interaction	Specifies requirement that containment accommodate FCI without excessive leakage	New RG needed
11	10CFR50 Appendix A General Design Criteria	GDC 50(c) Containment design basis-hydrogen combustion and detonation	Specifies requirement that containment accommodate hydrogen combustion or detonation without excessive leakage	New RG needed
12	10CFR50 Appendix A General Design Criteria	GDC 50(d) Containment design basis-melt attack	Specifies requirement that containment accommodate direct attack of molten corium without excessive leakage	New RG needed

	Regulation	Subpart	Description of Change	Ancillary Requirements
13	10CFR50 Appendix A General Design Criteria	GDC 50(e) Containment design basis-high pressure melt ejection	Specifies requirement that containment accommodate high pressure melt ejection without excessive leakage	New RG needed
14	10CFR50 Appendix A General Design Criteria	GDC 50(f) Containment design basis-corium-concrete interaction	Specifies requirement that containment accommodate corium-concrete interaction without excessive leakage or contamination of groundwater	New RG needed
15	10CFR50 Appendix A General Design Criteria	GDC 50(g) Containment design basis-pressurization by steam	Specifies requirement that containment accommodate pressurization by steam from decay heat without excessive leakage	New RG needed
16	10CFR50 Appendix A General Design Criteria	GDC 50(h) Containment design basis-elevated temperatures	Specifies requirement that containment accommodate elevated temperatures without excessive leakage or damage to key equipment	New RG needed

17	10CFR50 Appendix A General Design Criteria	GDC 51 Fracture prevention of containment pressure boundary	No change required	None
18	10CFR50 Appendix A General Design Criteria	GDC 52 Capability for containment leakage rate testing	No change required	None

	Regulation	Subpart	Description of Change	Ancillary Requirements
19	10CFR50 Appendix A General Design Criteria	GDC 53 Provisions for containment testing and inspection	No change required	None
20	10CFR50 Appendix A General Design Criteria	GDC 54 Piping systems penetrating containment	Simplify and make consistent with new GDC 58	New RG may be needed
21	10CFR50 Appendix A General Design Criteria	GDC 55 Reactor coolant pressure boundary penetrating containment	Simplify and make consistent with new GDC 58	New RG may be needed
22	10CFR50 Appendix A General Design Criteria	GDC 56 Primary containment isolation	Simplify and make consistent with new GDC 58	New RG may be needed
23	10CFR50 Appendix A General Design Criteria	GDC 57 Closed system isolation valves	Simplify and make consistent with new GDC 58	New RG may be needed
24	10CFR50 Appendix A General Design Criteria	New GDC 58 On-line monitoring of containment isolation status	Reduces likelihood of inadvertent bypass	New RG needed
25	10CFR50 Appendix A	New GDC 59 Protection of	Alternatively, revise GDC 2 to include structure	New RG needed

General Design Criteria	nuclear components against external threats	challenges from aircraft crashes, etc.

	Regulation	Subpart	Description of Change	Ancillary Requirements
26	10CFR50 Appendix A General Design Criteria	New GDC 59-A Assurance of containment integrity during shutdown	Ensure containment designs to permit emergency closure during shutdown operations	New RG needed
27	10CFR100 Reactor Site Criteria	NA	Siting criteria will be uncoupled from contain-ment design criteria in a separate staff program	New criteria/RG as appropriate
28	10CFR50 Appendix J Containment Leakage Testing	NA	Allowable leak rate as established in GDC 16; credit on-line monitoring per new GDC 58	New RG needed
29	10CFR50.44 Combustible Gas Control	NA	Superseded by GDC 50(c) in part	Delete
30	10 CFR 50.34(f) TMI-Related Requirements	NA	Superseded by new GDC 50	Delete