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EFFECTS ON NRC REGULATIONS AND REGULATORY GUIDES RESULTING FROM CONSIDERATION OF DEGRADED CORE (CLASS 9) ACCIDENTS

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ABSTRACT

This report describes the results of a review of the Code of Federal Regulations and certain Regulatory Guides for the possible impact of considering core damage accidents in the licensing process. It was found that a small number of the above documents would require change while many others may require some revision depending on the future interpretation of existing regulations as well as how the Nuclear Regulatory Commission proposes to include such accidents in the regulatory base.

It should be emphasized that the above documents represent only a fraction of the entire regulatory base and that this study represents a preliminary "first cut" assessment of including consideration of core damage accidents as part of the licensing process.

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PREFACE

The Core Damage Regulation Study was performed by Sandia National Laboratories (Sandia) and its subcontractor, International Energy Associates Limited (IEAL). This work was done for the Office of Standards Development, U.S. Nuclear Regulatory Commission. This report serves as a "first step" in the review of the regulatory base in order to determine the effects of including consideration of core damage accidents as part of the licensing process.

Sandia served as the prime contractor responsible for the overall report content and technical direction of the program. In particular, much of the work reported in Section 2 of this report was performed by Sandia with additional input provided by IEAL. The review of the Code of Federal Regulations and the Regulatory Guides (Sections 3, 4, and 5) was performed by IEAL. As part of that review effort, the authors wish to acknowledge the contribution of Edward Kurdziel of IEAL who provided technical support to the review process.

EXECUTIVE SUMMARY

This study is intended to be a "first cut" assessment of the possible impact of including consideration of core damage accidents in the licensing process. The objective of this study is to determine the effects on NRC regulations and regulatory guides resulting from such consideration. It is anticipated that the results of this work may provide input for the upcoming hearings regarding possible "degraded core rulemaking."

It was found that certain sections of the regulations and regulatory guides would require change in order to be consistent with current knowledge and interpretations regarding core damage accidents. The documents which appear most likely to change are those that relate to current assumptions and analyses for loss of coolant accidents (LOCAs).

During this effort, it became obvious that a detailed definition of the degraded core accident is necessary in order for industry and the NRC to be able to properly address design requirements for mitigating such accidents if they are included in the licensing process.

Finally, this work is being extended to include consideration of "core melt" accidents which will be the subject of a future NUREG/CR report.

1.0 INTRODUCTION

The accident at the Three Mile Island, Unit 2, nuclear power plant on March 28, 1979, resulted in damage to the reactor core with an associated release of radioactive material to the primary coolant system and generation of hydrogen from fuel cladding-water reaction "...well in excess of the amounts required to be assumed for design purposes by the current Commission regulations."[1] As a result, the U.S. Nuclear Regulatory Commission (NRC) has initiated a long-term rulemaking to "...consider to what extent, if any, nuclear power plants should be designed to deal effectively with degraded core and core melt accidents."[1]

In support of NRC's rule-making on degraded core accidents, Sandia National Laboratories requested that International Energy Associates Limited (IEAL) review and analyze appropriate sections of Title 10 of the Code of Federal Regulations (CFR) and the NRC regulatory guides to determine to what extent these portions of the NRC regulatory base would have to be changed to include degraded core accidents in the licensing process. It is not now possible nor desirable to identify in detail how the various parts of the regulations and regulatory guides might be changed, but only to determine if they should be changed and, if so, why such a change might be required. Many changes in the regulations or guides could be deemed desirable in the event of a degraded core rule-making. This study identifies only those changes that would be mandated by direct conflict of existing regulations with the concept of a degraded core rule or its basis.

The review of the regulations focused primarily on 10 CFR Parts 20, 50, 55, and 100, although other Parts were reviewed and found to be potentially affected by inclusion of degraded core rules in NRC regulations (e.g., 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements"). Furthermore, only the Division 1 Regulatory Guides that deal with "Power Reactors" and certain other Division 5 and 7 Regulatory Guides were reviewed. In Sections 3 and 4 of this report, regulations and regulatory guides are not listed if they were not to be impacted by consideration of degraded cores. The Standard Review Plan and other design documents and standards for nuclear power plants were not included in the review.

2.0 BACKGROUND

The scope of work for this study, as described in the introduction to this report, was insufficient by itself to allow direct identification of regulations and guides which might require modification. Discussions and analyses conducted by IEAL and Sandia National Laboratories during the investigation sufficiently clarified an interpretation of the accident category for the purpose of identifying affected regulations. This section of the report discusses the structuring which was performed, as well as its limitations.

2.1 INTERPRETATION OF CORE DAMAGE ACCIDENT

The assumptions and guidance followed in identifying affected regulations and guides are described in the following two sections.

2.1.1 Continuum of Accidents and Regulatory Approaches

There is no single "core damage accident" which naturally stands out as a logical reference basis for change in the regulations and guidance. Instead, there is a continuum of accidents within which all possible core damage accidents will fall. The ultimate changes in the regulations will depend on the choice of a reference case (accident) along this continuum. The relevant parameters describing the core condition, the mechanisms of the accident, and the regulatory approach to these accidents all may vary in a wide range. This will affect the modifications required of existing regulations and guides.

The parameters describing the core condition following a degraded core accident are perhaps the most basic of these factors. They include the condition of the fuel cladding, the condition of the fuel pellets, and the core geometry. A staff report from the President's Commission on the Accident at Three Mile Island[2] concluded that, in the case of TMI-2, all the design limits for a loss-of-coolant accident (LOCA) were exceeded, and some changes in core geometry had occured. Although estimates of core parameters exist for the accident at TMI, it must be remembered that the available information on the extent and nature of damage at the site is still equivocal. More significantly, the accident was only a single point in a range of such accidents for which remedies may vary significantly.

The possible mechanisms of degraded core accident initiation would also vary from case to case. A mechanistic accident analysis could conceivably yield detail or even bounds on the core condition discussed previously. More importantly, such an analysis would be required to supply information regarding system availability for mitigation of the damaged core sufficient to prevent core melt. (Clearly a LOCA which went beyond the design basis due to partial failure of low pressure injection may not be able to rely upon this system to provide long-term coolant makeup in the mitigation of that accident.) Without this type of data, specific conclusions cannot be drawn regarding means for mitigating a damaged core accident. Reasonable examination of tradeoffs between prevention and mitigation for specific accident sequences is also impossible without information on the genesis of the accident.

Finally, a variety of possible regulatory approaches to this issue is possible. Each approach would entail varying information requirements on the part of NRC, hence would imply different changes in regulations and guides. NRC must, for instance, determine how the potential new requirements will be added. One possibility is the postulation of a new accident or group of accidents to be added to the safety design basis for each plant. Alternatively, specific systems intended to mitigate certain conditions due to a damaged core could be mandated. Similarly, regulatory choices between mechanistic and non-mechanistic accident specifications condition the required modifications.

The level of regulatory detail at which NRC desires change also affects the findings of this study. Regulations are generally less detailed than regulatory guides, which, in turn, are usually less specific than industry standards. License conditions are more specific than all of the above. Since not all regulations or guides are at uniform levels of detail, NRC's preferences with respect to this issue will determine the scope of modifications required, the documents in which those changes appear, the clarity and uniformity of the regulatory requirements, and the balance of effort between rule-making and license-specific modifications.

2.1.2 Assumptions and Approach

Given the variety of possible core damage accidents and regulatory approaches as discussed above, additional assumptions were required to identify regulatory changes. One approach could have been explicit construction of specific accident sequences and resulting core damage. Alternatively, data from TMI could have been used. Instead, this study constructs a framework for examination of the potential regulatory changes in the most general manner possible. This choice was dictated by available time and level of effort, and by the intended use of the results. Ideally, more specificity can eventually be achieved with a clear, unambiguous description of the core damage accident to be considered in a preferred regulatory framework.

To retain broad applicability of the results, a qualitative approach was followed to identify regulations and guides which might require change. That is, a set of required functions and a list of reactor environmental conditions in the event of a core damage accident were identified, but no attempt was made to establish strict specifications or limits for the performance of the functions, or for the environmental conditions. (These functions and environmental conditions are discussed in Sections 2.2 and 2.4.) Particular reactor systems required to achieve the functions were also not identified since these may vary depending on plant type and the age of the plant. The only specific limit assumed was mitigation of the accident prior to melt-through of the reactor pressure vessel. This limit was established for the purposes of this study to distinguish between a degraded core accident and a melted core accident, the latter to be the focus of a separate effort. The review of regulatory documents used the general functions and conditions described in Sections 2.2 and 2.4 as guidance in identifying possible modifications.

Since the study assumed no specific values for accident conditions or system performance, each of these factors could adopt values which would depend on where the accident fell within the continuum of possibilities. This leads to a "conservative" approach in that some regulatory modifications identified would result only from a damaged core accident or a regulatory approach assuming worst-case conditions. For instance, in some cases, analysis of a design basis LOCA could demonstrate temperatures in a given system as great as those in a similar LOCA with core damage. Nonetheless, the potentially greater temperatures in a damaged core accident, and the generalized approach, require the assumption of a more stringent temperature limit requirement than that currently bounded by a design basis LOCA; or, at least, preparation of a more detailed analysis of this possibility. The result of this approach is a substantial body of regulations and guides which could require changes depending on the detailed basis for an eventual rule-making on degraded cores. All the modifications logically stem, however, from the guidelines on required safety functions and accident environmental conditions.

During the review it became evident that the affected regulatory documents could be separated into three categories. The subject of each category supports modification of existing requirements due to a degraded core condition. Viewing the changes in this light also helps to organize the results. Category 1 changes are those necessitated by a degraded core due to new conditions which must be considered for emergency equipment and accident analyses. In this case, the degraded core represents an additional accident condition for the plant. Category 2 modifications are required since normal (not necessarily safety) equipment may be used or called upon to function under adverse environmental conditions following a degraded core accident. This could include systems such as the chemical volume control system (CVCS), the residual heat removal system (RHR), steam generators, and the auxiliary feedwater, among others. In the specific identification of possible regulatory and guide changes in Chapters 3.0 and 4.0 the category expressing the rationale for each change is contained in parentheses following the discussion pertaining to that section.

In addition to the Category 1 and 2 changes, another category of possible changes to the regulatory documents, due to addressing degraded core accidents (DCAs), was considered. This category consists of modifications that reflect that the degraded core accident results in a change of plant condition over a long term, during which the potential probabilities and consequences of additional accidents have not been analyzed. That is, during the long term following a DCA, subsequent external events such as an earthquake or a tornado could be detrimental to the stability of the safe shutdown of the plant. Also, later failures of equipment being used to maintain the plant in a stable condition could have significant effects on the continued safety of the cleanup operations of the plant.

After considerable discussion, however, it was decided through the mutual agreement of IEAL, Sandia National Laboratories, and the NRC, not to include this category in the analysis of the regulations and regulatory guides. The principal reason for supporting this decision is that, traditionally, post accident conditions following (design basis) accidents (including design basis LOCAs) have not been addressed within 10 CFR, and it was felt that there was little justification for including coverage of only post-degraded core accident conditions in the regulatory documents.

2.2 REQUIRED SAFETY FUNCTIONS

The following general functions would be required in the event of a core damage accident. They would provide for short-term stabilization of reactor condition (prevention of further damage/core melt), mitigation of radiological hazard potential, and long-term recovery. These functions have been used as guidelines in the identification of requirements for which some adaptation would be implied by degraded core rule-making.

Reactor Subcriticality. Potential core geometry changes and loss of control functions due to the accident create the need to assure that shutdown can be achieved and maintained under degraded core accident conditions. Failure to maintain this function could lead to overpower and core melt. In general, the requirement will be in terms of maintaining subcriticality. If an anticipated transient without scram (ATWS) is considered as a potential cause of the degraded core accident, then subcriticality may need to be achieved, if possible, under degraded core conditions.

Primary System Pressure Control. Transient overpressurization could compromise the integrity of the primary system. Thus control of overpressurization is required. Pressure control for the prevention of low primary system pressure is also needed, to maintain subcooling.

Water Inventory. Effective heat transfer from the core requires that primary water inventory be maintained or recovered quickly in the event of a large LOCA.

Primary System Heat Removal. Capability to remove heat from the primary system is required. This may be accomplished through secondary side heat transfer, through the residual heat removal system, or through heat and mass transfer from the primary system to containment. In the latter case, containment heat removal must be adequate.

Containment Integrity. This function serves a dual purpose. Proper containment function may be necessary for heat removal and for proper Emergency Core Cooling System (ECCS) function. It also is required in the event of a damaged core to control radioactive releases to the outside environment.

Radioactivity Control. The potentially large releases of radioactivity to containment in the event of a damaged core require control to limit releases to the outside environment, as well as to allow eventual cleanup.

Cleanup/Decommissioning. The magnitude of damage to the core and possibly to other reactor systems inherent in a degraded core accident requires consideration of clean-up operations or even decommissioning in order to be consistent with the new requirements. Emergency planning, procedures, equipment, and financing may be required to handle this phase of accident recovery with minimal hazard to the public.

2.3 GENERAL DESCRIPTION OF DEGRADED CORE ACCIDENT

It is essential to understand the anticipated reactor environmental conditions during and after a degraded core accident. These conditions help determine performance criteria for the required safety functions, and define the hostile environment within which equipment will be forced to operate during such an accident. In order to derive these environmental conditions, the phenomena which lead to and result from core damage must be at least superficially understood.

Many details of these phenomena are not yet well understood and, in fact, probably vary depending on the initiating event, subsequent failures, and differences in basic design, particularly between the PWR and BWR. However, from the literature reviewed for this study (3,4,5,6,7,8) certain considerations are apparent which could affect the performance of the functions necessary to mitigate a core damage accident before gross melting occurs. In this section, a brief generic sequence of events during a degraded core accident is described. The following section then describes the environmental conditions identified in this sequence which must be addressed in degraded core accident considerations.

It is assumed that core damage would not result unless there is inadequate cooling of at least some portion of the core. (An exception to this could be direct structural damage to the fuel initiated by an event such as an earthquake. This is considered much less likely, however, and is not discussed here.) High temperatures would, then, initially exist in the fuel region and the core would probably be at least partially uncovered. These high fuel temperatures and the existence of a water-steam mixture, if unchecked, would cause oxidation of the zircaloy cladding. The oxidation process is exothermic, thus adding to the heat in the core region. Therefore, cooling the fuel becomes more difficult, and cladding oxidation is perpetuated. The reaction also generates hydrogen gas which may add to the pressure increase in the primary system (unless a LOCA exists, in which case hydrogen can escape to the containment atmosphere as it is generated). Clad failure would eventually occur, probably by fracturing of the oxygen embrittled cladding and spallation of ZrO_2 from the clad surface. If sufficient clad failure occurs, some deformation of the fuel could begin which could hamper the coolability of the core. Fission product release from the fuel would be probable from the points of cladding failure. The material released would include gases from the fuel-clad gap, and volatilized solids due to the high fuel temperatures. These events in the region of the core, within the primary system, could be reflected in conditions within reactor containment as well, depending on such factors as pressure relief to the containment, leakage rates from the primary system, etc. Failure or bypass of containment, allowing release of radioactivity, steam, and hydrogen gas could occur, depending on the success of containment isolation, failures in such equipment as the steam generator tubes in a PWR, failures or leaks from the equipment which may be outside containment but in "communication" with the primary system and/or containment environment, and the leakage rate of containment.

Additional problems exist due to possible steam or hydrogen explosions as their quantities become larger in an atmosphere of rising temperatures. There are many uncertainties in the literature regarding the probability of such explosions. However, it appears that the trend of current thinking suggests that steam explosions, particularly in the primary vessel, are much less likely than originally thought. Hydrogen gas formation would certainly have to be dealt with, however, it is apparent that the concentration of the gas may not reach critical values for detonation in some containment designs. A variety of ways have been suggested to deal with hydrogen build-up.

2.4 ANTICIPATED ACCIDENT ENVIRONMENT

From the general accident description above, the possible nature of the reactor environment due to such accidents can be postulated. It is the accident environment which could change the nature of and the demand for required safety functions, since none of these functions, by themselves, are uniquely required by a degraded core accident. The distinct environment could affect the operation of safety equipment for accident mitigation, as well as the normal, unfailed equipment surviving the accident. The accident environment may also affect the magnitude and types of consequences to the public and to plant workers. The significant features of the core damage accident environment are high temperatures, high pressure, noncondensible gas formation, steam generation, high radiation levels, and corrosive materials. Each of these topics is discussed briefly below.

High Temperatures. Other than an earthquake, the damaged core has resulted from fuel temperatures higher than any expected in the safety design basis. Metal/water reaction of the zircaloy cladding is then accelerated, and this exothermic reaction exacerbates the already excessive temperature in and around the core. Other factors in the accident condition may also lead to detrimental heat transfer and resultant high temperature. For instance, cooling channels blocked by eroded core and reactor internals may lead to localized hot spots. High temperatures in the containment could also affect operation of consequence limiting equipment used to reduce the risk from such accidents. In general, without knowing the genesis of the accident, we cannot say that peak temperatures will be less than those for any other accident condition, since a LOCA or any other transient could have caused the core damage. On the other hand, the above considerations indicate that without further analysis it is logical to assume that degraded core conditions could lead to more severe temperature conditions than would other accidents.

High Pressure. High pressures in the primary, secondary, and containment systems can result directly from the higher temperatures discussed above. In addition, generation of non-condensible gases increase primary system pressures and ultimately, those of containment. Steam generation also may add to containment pressurization, although this effect may not be significantly greater than any severe LOCA conditions (without core damage). Combustion of hydrogen can also lead to increases in peak containment pressure (and temperatures) for damaged core accidents.

Non-Condensible Gas Formation. A particular characteristic of a damaged core accident is the formation of quantities of non-condensible gases substantially in excess of those generated in any non-degraded core accident. In particular, hydrogen gas is generated as a result of oxidation of the Zircaloy fuel cladding at the high fuel temperatures experienced during such an accident. These gases can gather in the primary system, leading to overpressurization and to impairment of natural circulation. These gases in containment can lead to gradual overpressurization, or to sudden loads due to detonation in the presence of oxygen. Unknown impacts on instrumentation and control may also be a factor.

Steam Generation. Steam generation during the accident may also lead to overpressurization and to high humidity in containment. The latter could have an adverse effect on equipment not qualified for that environment. Steam in the primary system could reduce heat transfer from the core. In most respects, however, it appears likely that steam generation problems may not be more severe for these accidents than for severe LOCA scenarios.

High Radiation Levels. Damaged core accidents are distinguishable from others in that the integrity of at least one fission product barrier, i.e., the fuel cladding, is compromised to an extent greater than that for any accident within the current design basis. This, therefore, will lead to higher levels of radioactive material in the primary system than for current design basis accidents. This high level of radioactivity would likely end up in containment. Contamination of containment atmosphere and surfaces would result. Depending on the accident causing the core damage, other systems (e.g., secondary, auxiliary, outside environment) could be contaminated as well. The high activity levels can have an effect on instrumentation, personnel access and radiation exposure, operator actions, releases to outside environment, and reactor materials. It will also impact cleanup and decommissioning decisions and waste disposal requirements.

Corrosive Materials. High levels of boric acid to maintain subcriticality, combined with changed water chemistry, and flooding of parts of the reactor and containment not designed for this environment, all may have an uncertain effect on reactor materials over the course of recovery.

Others. The above environmental conditions will most likely exist in combination. The synergistic effects on materials of temperature, pressure, radiation, corrosive fluids, and humidity are not necessarily known for all reactor materials.

Previous discussion centered on possible abnormal levels of various environmental parameters. Another significant factor may be the rate of change in some of those variables. The rate of temperature increase, for instance, and the resulting thermal shock to reactor materials (particularly the pressure vessel) may be significant. This factor, again, may not be qualitatively distinct from that under LOCA conditions, but the more severe scenario could involve a larger magnitude effect.

Finally, duration of the accident is an important consideration. The high levels of some of these factors (e.g., radioactivity) will tend to lengthen the period required for ultimate recovery from the accident. Simultaneously, the reactor systems will be exposed to high levels of some of the environmental parameters over the extended period.

2.5 LIMITATIONS ON SCOPE

Accident Definition. As indicated previously, there was a high degree of uncertainty in the accident definition and in the regulatory framework intended by the statement of work. These uncertainties influenced the approach adopted, and the degree of detail achieved in the results.

This limitation also precluded examination of the precursors and mechanisms of the accident necessary to define completely the plant state at any given time. Knowing how to design (or what to require) to mitigate the core damage accident is virtually impossible without knowledge of the plant state. Without such examination, it is very difficult to compare measures for accident prevention vs. mitigation.

Documents Reviewed. The scope of this effort was limited to consideration of regulations (10 CFR) and Regulatory Guides. In particular, it was focused on 10 CFR Parts 20, 50, 55, and 100, and Division 1 Regulatory Guides. For the most part, these documents are quite general. More specific information is contained in industry standards referenced by the guides, and in licensing documents (e.g., technical specifications, Standard Review Plan, safety analysis reports, etc.). Thus, the magnitude of regulatory structure which would be affected in the consideration of degraded cores should not be judged directly from this effort. Furthermore, the nature of the changes may also be sensitive to examination of other documents. For instance, ECCS requirements are a source of many of the possible modifications cited in this investigation. This could reflect the greater degree of detail with which ECCS and LOCA are treated in the regulations themselves, compared with other accidents.

Nature of Cited Sections; New Requirements. The review was directed toward those existing regulations or guides which analysis indicated required modification. The criterion used was that a change would be needed if the existing document was in conflict with a degraded core accident requirement. For instance, hydrogen generation was specified in some places in much smaller quantities than a degraded core would imply.

A less dramatic instance is where LOCA is specifically indicated as an accident to be treated. Strictly speaking, this reference could remain but it was cited on the basis that the wording implied or stated that the (current design basis) LOCA was the most severe design basis condition. In some cases, marginal changes were indicated which "conflicted" only with the desired consistency or clarity of exposition of degraded core requirements.

It is most important to note that the study specifically excluded consideration of new documents. The changes considered were solely those required or desirable for resolution of existing requirements with damaged core considerations. In some cases, it was apparent that other classifications or additional guidance would be desirable, but such additions were considered outside the scope of this effort. The nature of any such new requirements should be decided when greater detail is available regarding the postulated accident to be mitigated, and a regulatory approach is chosen.

Status of Changing Requirements. The accident at Three Mile Island was more than just an example of a degraded core accident. It served as a focal point for a number of reactor safety concerns in addition to degraded core accidents. Examples include attention to the role of the operator in reactor safety, to personnel selection and training, to the phenomenology and mitigation of small vs. large break LOCA, and more. This review examined only those issues related specifically to the degraded core accident.

Since there has been a great deal of flux in requirements since TMI, Chapter 4.0 specifies the version of each regulatory guide examined. The 1980 issue of 10 CFR was reviewed. No documentation was examined which was not formally in the public domain, at least for comment.

3.0 EFFECT OF DEGRADED CORE ACCIDENTS ON THE REGULATIONS

This section presents a summary of the revisions that might be required in the Code of Federal Regulations (Title 10) as a result of considering degraded core accidents in the NRC regulatory process. These revisions were identified during a review of all Parts of Title 10 CFR (with particular emphasis on Parts 20, 50, 55, and 100) using as guidance the background structure, assumptions, and limitations discussed in Section 2.

PART 20 - STANDARDS FOR PROTECTION AGAINST RADIATION

The regulations in this Part establish standards for protection against radiation hazards arising out of licensed activities including operation of nuclear power plants.

This Part has traditionally applied only to release standards, permissible dose levels, and designation of restricted and unrestricted areas during routine operations. Standards covering similar topics have not been developed for accident or emergency conditions. Therefore, consideration may be given to modifying Part 20 to include requirements related to increased radiation levels within and around the plant due to degraded core conditions. It is noted that preliminary NRC staff positions, as expressed in the Advanced Notice of Proposed Rulemaking for Part 20,[9] include consideration of standards covering accident and emergency situations. Clearly, the opportunity exists to include degraded core accidents in the set of accidents to be covered in revisions to Part 20. (1,2)

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

This Part contains the major portion of NRC regulations pertaining to the design, construction, and operation of a nuclear power plant under conditions ranging from "normal" to "emergency." Therefore, as would be expected, Part 50 contains many sections that would more than likely, and in some cases certainly, be changed if degraded core accidents are to be con ered. Following are section-by-section discussions of possible changes within Part 50.

§50.33 CONTENTS OF APPLICATIONS; GENERAL INFORMATION

§50.33 (f)

This section contains information regarding financial qualifications of applicants covering operation, shutdown, and maintenance of a plant.

A statement could be added to cover financial aspects of a degraded core condition and, in particular, the added cost of the resulting cleanup operations.

§50.34 CONTENTS OF APPLICATIONS; TECHNICAL INFORMATION

§50.34 (a) Preliminary Safety Analysis Report (PSAR)

This section contains information regarding the content required in a preliminary safety analysis report (PSAR). A discussion of a degraded core condition as one of the emergency or accident conditions might be included. More specific points are addressed in the following subsections.

§50.34 (a) (4)

This section requires preliminary analysis and evaluation of the design and performance of various structures, systems, and components of a nuclear reactor unit, including the ECCS, under normal and transient conditions, as well as accident conditions, specifically a LOCA.

These conditions may have to be expanded to include degraded core conditions in addition to or in place of a postulated LOCA, due to the potentially more severe accident environment implied by such an accident. (1)

§50.34 (a) (10)

Part of this section addresses the applicants' plans for coping with emergencies.

It may be desirable to specify a degraded core condition and the followup cleanup procedures as an emergency. If this is the case, a change in this regulation may be needed. However, significant changes are being proposed for Appendix E, "Emergency Plans For Production and Utilization Facilities," and emergency planning for degraded core accidents might well be included in that section. (1)

§50.34 (b) Final Safety Analysis Report (FSAR)

This section contains information regarding the required content of a final safety analysis report (FSAR), particularly the design bases and limits on operation. A discussion of a degraded core condition might be included as one of the emergency or accident conditions. More specific points are addressed in the following subsections.

§50.34 (b) (4)

Similar section as §50.34 (a) (4) relating to the FSAR. (1)

§50.34 (b) (6)

Similar section as §50.34 (a) (10) relating to emergencies. (1)

§50.36 Technical Specifications

This section deals with the requirements for the technical specifications for a specific plant.

These technical specifications do not now address degraded core conditions. Technical specifications might be revised to deal with the prevention of accident sequences leading to degraded core conditions. (1)

§50.44 STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT WATER COOLED REACTORS

§50.44 (a)

This section states that each BWR and PWR will be able to control hydrogen gas that may be generated in a postulated LOCA.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA. This change may be needed because of the potentially higher production of hydrogen gas resulting from a degraded core accident. (1)

§50.44 (b)

This section states that each BWR and PWR will be able to measure the hydrogen concentration in the containment, ensure a mixed atmosphere, and control combustible gas concentrations following a postulated LOCA.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA. This change may be needed because of the potentially higher production of hydrogen gas resulting from a degraded core. (1)

§50.44 (c)

This section states that each BWR and PWR will either be able to guard against a hydrogen-oxygen recombination or withstand a hydrogen explosion following a postulated LOCA.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA. This change in the regulation may be needed because of the potentially higher production of hydrogen gas resulting from a degraded core. (1)

§50.44 (d) (1)

This section contains assumptions regarding the amount of hydrogen produced from a metal-water reaction following a postulated LOCA in plants which are in compliance with §50.46 (b). The section states the reaction is assumed to occur because of degradation, but not total failure, of emergency core cooling. Also, a time period of two minutes following the postulated LOCA is stipulated as the time of occurrence of the metal-water reaction.

This section will have to change on a number of counts. A degraded core accident by its very nature would exceed the limits on degradation of the core specified in this section. The two-minute period for a metal-water reaction may no longer be valid since the reaction might be sustained for longer periods of time under degraded core conditions. (1)

§50.44 (d) (2)

This section applies to those facilities not covered under §50.44 (d) (1). It states that the amount of hydrogen generated by metal-water reaction is assumed to result from the reaction of 5% of the mass of metal in the cladding cylinders surrounding the fuel.

A degraded core condition could likely result in a larger amount of metal reacting. This regulation must change to reflect the potential for more extensive metal-water reaction. (1)

§50.44(e)

This section deals with combustible gas control for certain plants. It states that the primary means for controlling combustible gases following a LOCA in those plants shall be by a combustible gas control system, not by purging or repressurization. Such a system must ensure that control of such gases does not result in a significant release from containment.

This regulation could be changed in order to extend the requirements to include combustible gas control for a degraded core accident. A degraded core accident (DCA) would produce greater volumes of combustible gases than would a design basis LOCA, which the control system would be required to handle. (1)

§50.44 (f,g)

These sections deal with the conditions under which purging systems would be allowed as the sole means for combustible gas control.

If a degraded core is used as the limiting condition, higher radiation levels and larger quantities of combustible gas could exist. Thus the rule might be changed to require additional means other than purging systems for combustible gas control without exception, in order to continue to meet various offsite boundary criteria. (1)

§50.44 (h) (1)

This section defines degradation of the ECCS for purposes of design of the combustible gas control system.

A change may be needed to increase the amount of degradation that is assumed for the fuel due to a degraded core condition. The assumption of no core melting might be changed to include partial core melting, but in such a form as to be useful for design purposes. (1)

§50.44 (h) (2)

This section defines a combustible gas control system as a system that operates after a LOCA to maintain concentrations of gases in the containment below flammability limits.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA. A change may be needed because a degraded core condition would result in potentially higher gas concentrations to be controlled, hence the LOCA would no longer adequately bound the combustible gas production for design analysis. (1)

§50.46 ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT WATER COOLED NUCLEAR POWER REACTORS

In the discussions that follow, it is recognized that no change may be desired to the existing ECCS acceptance criteria as they relate to current design basis LOCA conditions. Instead, additional criteria may be appropriate for ECCS performance (or the performance of yet another accident mitigating system) under degraded core conditions. A fundamental issue that must be resolved prior to degraded core rulemaking is how the degraded core accident is to be included in the design basis and associated safety analysis framework.

§50.46 (a)

This section addresses the acceptance criteria for emergency core cooling systems for nuclear reactors, including ECCS performance under a range of postulated loss-of-coolant accidents.

There may be a change required in order to include ECCS performance under degraded core conditions. The environmental conditions, such as temperature and pressure, could be more severe for degraded core conditions than for a postulated LOCA. Moreover, the performance criteria for the ECCS might be more demanding if it is required to function to mitigate a degraded core. (1)

§50.46 (b) (1) Peak Cladding Temperature

This section states that the maximum fuel element cladding temperature shall not exceed 2200°F.

If a degraded core condition is assumed, there may be a change in this regulation. The peak cladding temperature of 2200°F was chosen to limit excessive degradation of the cladding. However, in a degraded core, the cladding has either oxidized or melted, and the 2200°F limit may have been surpassed. Thus, a change may be necessary in the temperature limit, or secondary criteria should be formulated to cover a degraded core condition. For instance, current limits could be used as a design criteria for ECCS performance to avoid fuel degradation. If core degradation and these limits were surpassed, the ECCS could be required to reduce excessive temperatures, and maintain them, below some new limits within some period of time. (1)

§50.46 (b) (2) Maximum Cladding Oxidation

This section states that the maximum cladding oxidation shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

There might be a change in this section similar to the one above, if degraded core conditions occur. If degraded core conditions exist, there could be much more oxidation than is assumed in the rule, as learned from TMI. Thus, another limit on maximum oxidation must be set, or secondary criteria must be formulated to cover degraded core conditions. (1)

§50.46 (b) (3) Maximum Hydrogen Generation

This section states that the total amount of hydrogen generated from the chemical reaction of the cladding with water would not exceed .01 times the amount if all the metal reacted.

This section may change if degraded core conditions are assumed. With a degraded core there could be more extensive metal-water reaction and more hydrogen production can occur. (1)

§50.46 (b) (4) Coolable Geometry

This section states that the changes in core geometry shall be such that the core remains amenable to cooling.

If a degraded core condition exists, the core may no longer be amenable to cooling with the existing ECCS system, due to crumbling of the core (from loss of cladding integrity) or partial melting of the fuel. There may be a change necessary to include this condition. (1)

§50.46 (c) (2)

This section gives information on an evaluation model used for assessing the behavior of the reactor system during a postulated LOCA.

If a degraded core condition is now assumed to be the most severe reactor condition, an evaluation model for the behavior of the reactor system during a degraded core situation should be developed. (1)

§50.55 CONDITIONS OF CONSTRUCTION PERMITS

§50.55a Codes and Standards

This section defines the industry codes and standards to which the structures, systems, and components of a plant shall be designed, fabricated, erected, constructed, tested, and inspected.

The use of referenced codes needs to be examined in light of changes in temperature, pressure, and source terms due to a degraded core condition. If a reference is no longer applicable, or is inadequate, the reference in this section may need change. (1,2)

§50.82 APPLICATION FOR TERMINATION OF LICENSES

This section deals with application by a licensee to the Commission for authority to surrender a license voluntarily and to dismantle a facility and dispose of its component parts.

No detailed decommissioning regulations now exist. However, they may be particularly necessary in the case of a plant with a damaged core.

50 Appendix A General Design Criteria for Nuclear Power Plants Definitions and Explanations

The definition of a "degraded core" must be inserted into this section. This definition may affect the degree of change required in the subsequent criteria in this section. For example, if a degraded core accident were to be included, by definition, within the set of postulated accidents, then few if any changes would be required in the General Design Criteria. If this approach were used, then some of the changes we have indicated would not be necessary.

Criterion 4 - Environmental and missile design bases

This criterion states that all plants shall be designed for the environmental conditions associated with a range of plant states from normal operation to postulated LOCAs, and for dynamic effects, such as missiles.

A change may be necessary because the environmental conditions and the consequences of missiles might be more demanding for a degraded core accident and the subsequent condition than for a LOCA. (1,2)

Criterion 13 - Instrumentation and control

This criterion addresses instrumentation and control, which must be provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions.

Under degraded core conditions, instrumentation and control requirements may become more stringent, as it becomes more difficult to maintain the variables and systems within prescribed operating ranges. Higher limits for most environmental variables are possible, affecting the range of instruments. More stringent environmental qualification could be required, as could operating conditions for control equipment. (1,2)

Criterion 16 - Containment design

This criterion states that containment design shall provide a leak-tight barrier against radioactivity release during postulated accident conditions.

It may be desirable to specify a degraded core condition as a postulated accident. If this is not the case, this criterion may have to change so that the containment is designed for higher pressure, temperature, and radioactivity. (1)

Criterion 17 - Electric power systems

This criterion requires the provision of onsite and offsite electric power systems to permit functioning of structures, systems, and components important to safety. Capabilities of these systems in the event of a LOCA and the single failure criteria for these systems are stated. Safety functions to be performed by the electric power system are to provide capacity and capability to keep the fuel within "acceptable fuel design limits" and to ensure that the core can be cooled and containment integrity maintained during postulated accidents.

A degraded core condition could present greater challenges to performance and a more stringent demand for electric power systems. The criteria to be met by the systems might, therefore, change from LOCA to degraded core accident. Some of the requirements of this section, such as maintenance of fuel within design limits and single failure criteria, could be violated by a degraded core accident. Criterion 17 might be rewritten to show consistency with the possibility of a degraded core accident. (1,2)

Criterion 19 - Control room

This criterion describes the design bases for a control room at a nuclear power plant. The control room should provide a safe area from which one can operate the nuclear unit to a safe shutdown under accident conditions. LOCA conditions are specifically cited. Control capability outside the control room is specified.

This criterion may change if degraded core conditions are assumed. Protection against higher radiation levels associated with a degraded core might have to be required. "Safe shutdown" might require interpretation in the event of such an accident, since the long-term recovery/cleanup period does not correspond with the typical notion of "safe shutdown". Degraded core could be mentioned instead of, or in addition to LOCA conditions. Outside control room measurement and control capabilities could be established for degraded core accidents. (1,2)

Criterion 20 - Protection system functions

This criterion states that the protection system shall be designed to sense accident conditions and automatically initiate the appropriate safety systems so that acceptable fuel design limits will not be exceeded.

If a degraded core condition is assumed, the acceptable fuel design limits presumably would have been exceeded. Thus, this criterion may require change to be made consistent with the assumption that a degraded core condition might exist. Provision of additional

RPS functions in the event of a degraded core accident could be considered. Maintenance of those functional capabilities under the more severe conditions of a degraded core accident might be specified. (1)

Criteria 27, 28

These criteria deal with controlling the reactivity in the core.

These criteria may need to change because of a degraded core situation. A degraded core could present a different core geometry, which may impose new requirements for maintaining subcriticality. (1)

Criterion 31 - Fracture prevention of reactor coolant pressure boundary

This criterion states that the reactor coolant pressure boundary should be designed to assure that it will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

The design of the reactor coolant pressure boundary to withstand brittle behavior may have to change. Under degraded core conditions, there may be higher service temperatures and more severe stresses on the pressure boundary, thus affecting this criterion. In addition, compromise of the pressure boundary could have more severe consequences in the case of a damaged core accident than some other accident conditions, which might necessitate more stringent reliability criteria in this event. (1)

Criterion 33 - Reactor coolant makeup

This criterion states that a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary should be provided, so that acceptable fuel design limits are not exceeded.

If a degraded core condition is assumed, the acceptable fuel design limits presumably would have been exceeded. Thus, this criterion must be made consistent with the assumption that a degraded core condition might exist. Also, the reactor coolant makeup system may be required to work under more severe environmental conditions. This is a good example of the Category 2 modification discussed in Section 2.1.2. (2)

Criterion 34 - Residual heat removal

This criterion requires a residual heat removal system to remove decay heat so that acceptable fuel design limits are not exceeded.

If a degraded core condition is assumed, the acceptable fuel design limits would have been exceeded. Thus, this criterion must be made consistent with the assumption that a degraded core condition might exist. Also, the residual heat removal system may be required to work under more severe environmental conditions. (2)

Criterion 35 - Emergency core cooling

This criterion requires an emergency core cooling system (ECCS). The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate sufficient to prevent extensive core damage.

The design criteria of the ECCS could change under degraded core conditions because the ECCS may have to perform under conditions that include fuel and cladding damage and their effects as well as higher percentages and rates of metal-water reaction and the subsequent hydrogen production in the primary system. Additional system functions to recover from a period of inadequate heat transfer (by existing criteria), and to maintain primary system heat removal over the long-term recovery period could be required. (1)

Criterion 38 - Containment heat removal

This criterion states that the containment heat removal system shall reduce rapidly the containment pressure and temperature following any loss-of-coolant accident, and maintain pressure and temperature at acceptably low levels.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA. The design bases for the containment heat removal system may change to include maintaining acceptable conditions under possibly higher peak pressure, temperature, and humidity conditions. (1,2)

Criterion 41 - Containment atmosphere cleanup

This criterion states that systems to control fission products, hydrogen, oxygen, and other substances released to the containment following a postulated LOCA will be provided.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA. There may be a change because the cleanup systems may have to control higher amounts of fission products, hydrogen, oxygen, and aerosols over longer periods following the initial accident. (1)

Criterion 44 - Cooling water

This criterion states that a system be provided to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink.

It may be desirable to specify a degraded core condition and the subsequent cleanup as an accident condition. A change may be necessary because the cooling water system may now be operating under different design conditions. Failure of isolation provisions, for instance, could lead to higher releases due to the greater radioactivity potentially present in reactor coolant. (1,2)

Criterion 50 - Containment design basis

This criterion states that the reactor containment structure shall be designed so that the structure can accommodate the calculated pressure and temperature conditions resulting from a loss-of-coolant accident.

It may be desirable to specify a degraded core condition in addition to or in place of a postulated LOCA.

The design bases for the containment may have to change to compensate for potentially higher pressures and temperatures associated with a degraded core. (1)

Criterion 51 - Fracture prevention of containmen pressure boundary

This criterion states that the containment pressure boundary should be designed to assure that it will behave in a nonbrittle manner and the probability of rapidly propagating failure is minimized, even under postulated accident conditions.

The design of the containment pressure boundary to withstand brittle behavior may have to change. Under degraded core conditions, there may be higher temperatures and pressures leading to greater stresses. Furthermore, possibly higher radiation levels within containment could lead to larger offsite releases if the containment pressure boundary is compromised. (1)

Criterion 64 - Monitoring radioactivity releases

This criterion states that means be provided for monitoring radioactivity that may be released as a result of various conditions, including postulated accidents.

It may be desirable to specify a degraded core condition and the subsequent cleanup as a design basis. There may be a change because monitoring may have to be accomplished for higher source terms under more severe environmental conditions. Monitoring points, their accessability, and hazard to personnel must be considered. (1,2)

Part 50 Appendix C A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits and Operating Licenses

This Appendix lays out the general kinds of financial data and other related information that will demonstrate the financial qualifications of the license applicant to carry out the activities for which the license is sought.

There may be a change in this Appendix due to a degraded core. Core damage imposes a large financial burden on an operating utility. Requirements might also change to necessitate financial ability to safely decommission a severely damaged reactor.

Part 50 Appendix E Emergency Plans for Production and Utilization Facilities

This Appendix specifies minimum requirements for emergency plans. Substantial changes are already being considered for this Appendix E.

Part 50 Appendix G Fracture Toughness Requirements

This Appendix specifies minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of water-cooled reactors.

The fracture toughness requirements may change due to the possibly higher pressures and temperatures, and thus changes in hypothetical loads and shocks, due to a degraded core condition.

Part 50 Appendix H Reactor Vessel Material Surveillance Program Requirements

The purpose of the material surveillance program required by this Appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and the thermal environment.

The potentially higher stresses due to a degraded core may necessitate increased surveillance requirements, in order to ensure greater reliability of the reactor vessel under increased maximum analyzed stresses and shocks.

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

This Appendix sets forth the containment leakage test requirements for primary reactor containments. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment.

There may have to be modifications in the testing criteria if degraded core accidents are to be considered. Test intervals, for instance, might be shortened to ensure greater leak-tight reliability which might be necessitated by the possible higher levels of contamination and higher pressures due to a degraded core situation. (1)

Part 50 Appendix K ECCS Evaluation Models

This Appendix sets forth evaluation models for the ECCS under various stages of a postulated LOCA.

There may be changes or additions necessary if evaluation models are to be developed for degraded core conditions in addition to or in place of a LOCA. This could involve modification of the existing Appendix to account for a degraded core condition. (1)

PART 55 - OPERATORS' LICENSES

The regulations in this Part establish procedures and criteria for the issuance of licenses to reactor operators.

There may have to be modifications or additions to this Part to require operator training for degraded core accidents and subsequent cleanup.

PART 100 - REACTOR SITE CRITERIA

§100.11 DETERMINATION OF EXCLUSION AREA, LOW POPULATION ZONE, AND POPULATION CENTER DISTANCE

This section sets forth the basis for the numerical values used for the analysis to derive an exclusion area, low population zone, and population center distance.

There may be a change in the amount of assumed fission product release in a degraded core condition as compared to Technical Information Document (TID) 14844 release calculations. This could affect acceptable siting for future plants. Part 100 is currently under revision.

4.0 EFFECT OF DEGRADED CORE ACCIDENTS ON DIVISION 1 REGULATORY GUIDES

The following discussion presents a summary of each affected Division 1 Regulatory Guide and the revisions that might be required as a result of degraded core considerations. In general, a possible revision would fall under one of the two categories defined in Section 2.1.2. They are:

> Category 1. Those changes necessitated by new conditions and requirements imposed on emergency equipment, and accounted for in accident analyses.
>
> Category 2. Modifications required to account for more severe requirements imposed on normal (not necessarily safety) equipment.

The appropriate general reasons are indicated in parentheses after each revision discussed below. A more specific discussion of the reason for each revision is also included. In some cases a regulatory guide simply indicates approval of a particular industry code or standard. Since codes and standards were not reviewed, the regulatory guides in these cases were not listed.

1.1 Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1). (12/1/70) This guide recommends that emergency core cooling and containment heat removal systems be designed such that adequate net positive suction head (NPSH) is available assuming maximum expected temperature of the pumped fluids and no increase in containment pressure from that prior to postulated loss-of-coolant accidents (LOCAs). If a degraded core results in higher temperatures in the pumped fluids, this guide might be revised to state that adequate NPSH must be available during a degraded core as well as a LOCA. (1)

1.2 Thermal Shock to Reactor Pressure Vessels (Safety Guide 2). (12/1/70) This guide addresses the need to ensure that the injection of cold ECCS water following a LOCA will not induce brittle failure of the reactor vessel. Because of potentially higher temperatures in the vessel, during a degraded core situation, which could cause more severe thermal shocks, it might be necessary to revise the guide to address ECCS injection during a degraded core as well as a LOCA. (1)

1.3 <u>Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors</u>. (6/74) This guide lists the acceptable assumptions for evaluating radioactive releases to be used in the analysis of a postulated LOCA at a BWR. Some assumptions are that 25% of the iodine and 100% of the noble gases are released to the containment atmosphere, and that the containment leaks at the leak rate incorporated in the technical specifications. These assumptions may have to be revised to reflect a degraded core situation. Analysis of fluid pathways to the outside environment should be examined to ensure adequacy given the potentially higher releases. In addition, the containment pressure during a degraded core situation might exceed that during a design basis LOCA, in which case the containment leak rate would be higher than currently specified. (1)

1.4 <u>Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors</u>. (6/74) This guide lists the acceptable assumptions for evaluating radioactive releases to be used in the analysis of a postulated LOCA at a PWR. The revisions for Regulatory Guide 1.3 also apply here. (1)

1.5 <u>Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)</u>. (3/10/71) This guide lists acceptable assumptions to use in the analysis of a BWR steam line break accident. The radioactivity in the coolant is assumed to be the maximum amount incorporated in the technical specifications provided that no further fuel failures are assumed to occur as a result of delays in valve closure. If a steam line break accident can lead to a degraded core situation, then this guide might be revised to state that a much larger quantity of radioactivity should be assumed to exist in the coolant, reflecting the significant fuel damage in a degraded core; or it must be assumed that isolation of the break occurs before significant fuel damage can occur. (1)

1.7 Control of Combustible Gas Concentrations In Containment Following a Loss-of-Coolant Accident. (11/78) This guide discusses acceptable ways to control H_2 concentration in the containment following a LOCA. The hydrogen generated by the metal-water reaction is assumed to equal the amount generated by reacting 5% of the zircaloy in the core. This amount would have to be revised upward to account for the larger quantity of zircaloy that might react in a degraded core situation. Also, the assumed fission product distribution model (50% of core halogens and 1% of solids are mixed with the coolant water) might have to be revised to reflect more fuel degradation. (1)

1.11 Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations. (3/10/71) This guide provides guidance on the design of instrument lines penetrating or connected to primary reactor containment. The guide recommends that lines be sized such that a postulated rupture of the line outside primary containment will result in offsite exposures substantially below 10 CFR 100 guidelines. If such an accident occurred during a degraded core situation the offsite exposures could be higher than normally calculated because of the potentially higher level of radioactivity in the released coolant. Therefore, the guide might be revised to state that offsite exposures from an instrument line break should be calculated with the assumption that the core is degraded. (1)

1.21 Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants. (6/74) This guide explains how to measure and report both normal and abnormal radioactive releases to the environment. The guide might be revised to recommend upper limits for measuring instruments so that radiation measurements during a degraded core situation remain on-scale. However, we understand that revisions of this type will probably be included in Regulatory Guide 1.97. (1,2)

1.23 Onsite Meteorologic Programs (Safety Guide 23). (2/17/72) This guide describes how to monitor meteorological conditions around the plant site in order to estimate potential doses to the public from radioactive releases. Degraded core accident conditions and the potentially higher amounts and types of releases may have to be included in this guide. (1,2)

1.33 Quality Assurance Program Requirements (Operation). (8/79) Appendix A of this guide lists typical safety-related activities that should be covered by written procedures. The Appendix may be revised by adding degraded core mitigation as an activity requiring written procedures. Refer to ANSI 18.7-1976/ANS 3.2. (1,2)

1.40 Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants. (3/16/73) This guide describes procedures for testing Class I motors in the containment under simulated LOCA environmental conditions. The guide might be changed to include simulation of degraded core environment conditions if they are considered to be more severe than LOCA conditions. Refer to IEEE Std. 334-1971. (1)

1.48 Design Limits and Loading Combinations for Seismic Category I Fluid System Components. (5/73) This guide provides acceptable design limits and appropriate combinations of loadings for the design of Seismic Category I fluid system components. If a degraded core creates a more severe loading on the fluid system components than a design basis LOCA or normal plant conditions, then the guide may have to be revised to impose this more severe loading in the design of certain classes of components. (1)

1.52 Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature (ESF) Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants. (3/78) This guide presents acceptable methods for design, testing, and maintenance of post-accident ESF atmospheric cleanup systems. The guide states that ESF cleanup systems should be designed for the environmental conditions of a design basis LOCA. Since some environmental conditions (e.g., maximum pressure, pressure surge, and airborne radiation levels) in a degraded core situation might be higher than in a design basis LOCA, the guide might be revised to recommend using degraded core environmental conditions for the design basis. The guide also recommends using a 30-day integrated radiation dose in the design of the adsorber section. Since a degraded core situation might extend beyond 30 days, the guide might be revised to use a longer time period in calculating the integrated dose. Refer to ANSI N509-1976, ANSI N510-1975, and ERDA 76-21. (1)

1.57 Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components. (6/73) This guide provides acceptable loading combinations to be used in the design of metal containments. One of the design loads is the force from a design basis LOCA. If a degraded core accident can cause more severe loads (perhaps because of an H_2 explosion) on the containment, then the guide should be revised to include the degraded core as well as the design basis LOCA in determining design loads. (1)

1.67 Installation of Overpressure Protection Devices. (10/73) This guide provides design guidelines for reactor coolant system relief valves. If a degraded situation imposes more severe loadings than normally assumed in the design of these valves, then the guide might be revised to recommend use of the more severe design loads. For example, liquid water was discharged frequently from the pressurizer relief valve during the TMI accident. This may have imposed severe loadings on the valve if the valve was only designed for steam discharge. Refer to ASME Code Case 1569. (1,2)

1.68 Initial Test Programs for Water-Cooled Reactor Power Plants. (8/78) This guide describes the initial test program to verify correct operation of plant systems before beginning full power operation. The guide might be revised to recommend that each normal and emergency system be tested to verify correct operation in a degraded core situation if a degraded core requires a different function or mode of operation for that system. (1,2)

1.68.1 Preoperational and Initial Startup Testing of Feedwater and Condensate Systems For Boiling Water Reactor Power Plants. (1/77) This guide describes initial testing for BWR condensate and feedwater systems. If these systems might be required during a degraded core situation, then the guide should be revised to recommend that they be tested to verify correct operation during a degraded core situation as well as during normal and transient conditions. (2)

1.70 Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants. (11/78) This guide describes the content of safety analysis reports. Various sections might need to be revised, as discussed below, because of degraded core considerations.

Section 2.4.11.6 addresses heat sink dependability. If a degraded core imposes more severe requirements on the heat sink than traditional design basis accidents, then this section might be revised to consider the degraded core state when analyzing heat sink dependability. (1)

Sections 3.5.1.1, 3.5.1.2, and 3.5.2 address sources of missiles and systems that must be protected from missiles. If a degraded core situation could be the source of new missiles (e.g., from a steam explosion within the vessel), then these sections should be revised to include these degraded core considerations. (1)

Section 3.6.1.3 addresses pipe whip effects from postulated ruptures inside and outside containment. If new or existing systems needed for degraded core mitigation require pipe whip protection, then this section should be revised to include these systems in the analysis. Also, if pipe ruptures can occur during a degraded core accident and result in the release of contaminated fluids, then the section should be revised to include that consequence in the analysis, especially with regard to control room habitability. (1)

Sections 3.8.1.3, 3.8.2.3, 3.8.3.3, and 3.8.4.3 address the design loads to use in the design of the containment and other seismic Category I structures. Since the loads from a degraded core accident may be more severe than those from a design basis LOCA, the sections might be revised to consider the degraded core loads in the containment design. (1)

Section 3.9.1.1 describes the design transients that must be considered in the design of mechanical systems and components. The section might be revised to include a degraded core accident as another design transient to consider. (1)

Sections 3.9.4.3, 3.9.5.2, and 3.9.5.3 address design loads for the control rod drives and reactor internals. The sections might be revised to include loads from a degraded core accident in the design basis. (1)

Sections 3.11.1 and 3.11.5 address the environmental conditions during design basis accidents. The sections might be revised to also include the environmental conditions during a degraded core accident. (1)

Section 4.3.1 discusses the design basis for nuclear design of the fuel and reactivity control systems. This section might be revised to include a degraded core as a design basis. For example, a degraded core might result in damaged control rods which could affect reactivity control. (2)

Sections 4.4.4.4 and 4.4.6 discuss core thermal response and instrumentation requirements. These sections might be revised to include core thermal response and instrumentation during a degraded core situation. (2)

Sections 5.2.2.1 and 5.2.2.2 discuss the design basis for primary coolant boundary safety and relief valves. The sections might be revised to include a degraded core situation as another design basis, because relief rates and pressure transients might be more severe. (1)

Section 5.4.2 deals with PWR steam generators and requires a discussion of the consequences of potential tube ruptures. It might be revised to include tube rupture probability and consequences during a degraded core condition. (1,2)

Sections 5.4.6.1, 5.4.7.1, 5.4.8.1, and 5.4.11.1 discuss the design basis for the Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR), BWR reactor water clean-up, and pressurizer relief discharge systems. If these systems are needed for degraded core mitigation, then the sections should be revised to include a degraded core in the design basis. In particular, additional shielding, filters, and operational procedures might be needed in these systems to handle highly contaminated water. (2)

Sections 6.2.1.1, 6.2.2.1, 6.2.3.1, 6.2.4.1, and 6.2.5.1 discuss the design basis for the containment, containment heat removal system, secondary containment, containment isolation system, and combustible gas control in containment. These sections might be revised to include the degraded core as a design basis. In particular, the degraded core might mean higher pressures, higher radioactive source terms, and higher hydrogen levels for the design basis. (1)

Sections 6.3.1 and 6.3.3 discuss the design basis and performance evaluations of the ECCS. These sections might be revised to consider the ability of the ECCS to mitigate a degraded core situation. In particular, higher core temperatures, a less coolable core geometry, and a significant volume of non-condensibles might exist during a degraded core, which could impose more severe demands on the design of the ECCS. (1)

Sections 6.4.1 and 6.4.4.1 discuss the design basis and design evaluations of the control room habitability systems. These might be revised to include a degraded core in the design basis because of the potentially higher radiation levels during a degraded core accident. (1)

Sections 6.5.1.1, 6.5.2.1, 6.5.3.1, 6.5.3.2, and 6.5.4.1 discuss the design basis for the ESF filter system, containment spray system, and fission product control system. A degraded core might result in higher radiation releases than the traditional design basis accidents. If so, these sections might be revised to include a degraded core in the design basis. (1)

Section 6.7.1 discusses the design basis for the main steam isolation valve leakage control system in a BWR. Since a degraded core might result in higher radiation levels in the steam that leaks, this section might be revised to include a degraded core in the design basis. (1)

Chapter 7 discusses instrumentation and control. An additional section might be added to describe instrumentation specifically designed to detect or monitor degraded core conditions. (1,2)

Section 8.3.1.2 discusses the operation of safety-related electrical components in the hostile environment of postulated accidents. This section might be revised to include the hostile environment of a degraded core situation, which could involve higher pressure, humidity, and radiation levels. (1)

Section 9.2 discusses auxiliary water systems. If any of these water systems are needed for degraded core clean-up, then the appropriate section should be revised to include relevant degraded core considerations, such as the increased contamination levels. (2)

Section 9.3.2 discusses the process sampling system. This section might be revised to include obtaining samples during a degraded core situation, which could be more difficult because of the higher radiation levels. (2)

Section 9.3.3 discusses the equipment and floor drainage system. This section might be revised to include a discussion of how to ensure proper operation of the drainage system so that unnecessary contamination does not occur during a degraded core situation. (2)

Sections 9.3.4.1 and 9.3.5.1 discuss the design basis for the chemical and volume control system (PWR) and standby liquid control system. These sections might be revised to include a degraded core in the design basis. In particular, the reactivity control requirements of the boron system might be affected by the possibility of damaged control rods in a degraded core and also the presence of highly radioactive primary coolant. (2)

Section 9.4.1.1 discusses the design basis for the control room ventilation system. This section might be revised to include the higher airborne radiation levels that could result from a degraded core. (2)

Sections 9.4.2.2, 9.4.3.1, 9.4.4.1, and 9.4.5.1 discuss the design basis for the ventilation systems in the spent fuel pool, auxiliary and radwaste buildings, turbine building, and ESF room. These sections might be revised to discuss the possibility of higher airborne radiation levels due to a degraded core. (2)

Section 10.4.7 discusses the condensate and feedwater systems. If these systems are needed for cooling during a degraded core situation, then this section should be revised to include a discussion of that mode of cooling. (2)

Section 10.4.9.3 is an evaluation of auxiliary feedwater system performance. If this system is needed to provide long-term cooling for a degraded core, then this section should be revised to include a discussion of that function, particularly in light of more stringent reactor environmental conditions. (1)

Sections 12.2.1 and 12.2.2 describe the radiation sources that are to be used as the basis for radiation protection design. These sections might be revised to include any higher radiation levels or additional radiation sources that might be present during a degraded core situation. (2)

Sections 13.5.2.1 and 13.5.2.2 discuss plant operating procedures. These sections could be revised to include procedures for recognizing and mitigating a degraded core. (1,2)

Chapter 15 contains description regarding the safety analysis of postulated accidents. A degraded core could be addressed in two ways in these accident analyses: (1) one of the traditional initiating events is assumed to progress to the point where the core is degraded (the analysis would show that the core can be contained and radiological consequences do not exceed guidelines); (2) the degraded core is considered to be a new design basis event (nonmechanistic). In either case, a new design basis radiological source term might be needed. Also, the degree of failure of containment isolation might be reevaluated. (1)

1.73 Qualification Tests of Electric Valve Operations Installed Inside the Containment of Nuclear Power Plants. (1/74) This guide addresses qualification tests of electric valve operations inside the containment. If environmental conditions during a degraded core are more severe than during a LOCA, then the guide should be revised to stipulate the more severe conditions for the tests. Refer to IEEE Std 382-1972. (1)

1.81 Shared Emergency And Shutdown Electric Systems For Multi-Unit Nuclear Power Plants. (1/75) This guide describes acceptable arrangements for sharing of onsite electric power systems at multi-unit plants. This guide should be revised to account for the fact that degraded core accidents are included as a possible worst case in terms of power drain on the onsite electric system. (1)

1.82 Sumps For Emergency Core Cooling And Containment Spray Systems. (6/74) This guide provides guidelines for designing containment sumps used for emergency core cooling and containment spray systems. A degraded core might result in increased radiation levels and debris in the sump water. The guide might be revised to include these considerations in the design basis. (1)

1.89 Qualification Of Class IE Equipment For Nuclear Power Plants. (11/74) This guide discusses procedures for qualifying Class IE equipment under accident conditions. If environmental conditions in a degraded core situation are more severe than in a design basis LOCA, then the guide might be revised to stipulate the more severe conditions. (1)

1.96 Design Of Main Steam Isolation Valve Leakage Control Systems For Boiling Water Reactor Nuclear Power Plants. (6/76) This guide describes requirements for a leakage control system (LCS) for BWR main steam isolation valves. This guide might be revised to consider the potentially higher radiological consequences of leakage, and the potentially more severe environmental conditions for LCS operation, that might result from a degraded core. (1,2)

1.97 Instrumentation For Light-Water-Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident. (12/79) This guide describes instrumentation needed to provide the operator with sufficient information to follow the course of an accident. This guide appears to address degraded cores already. For example, it refers to "conditions that have degraded beyond the design bases". Refer to ANS-4.5. A revision to this guide has just recently been issued which reflects TMI experience. (1) (1)

1.105 Instrument Setpoints. (11/76) This guide can be used in the selection of required instrument accuracy and the settings used to initiate automatic protective actions and alarms. This guide might be revised to state that instrument accuracy, range, and setpoints must be adequate to deal with a degraded core situation. (1)

1.106 Thermal Overload Protection For Electric Motors On Motor-Operated Valves. (3/77) This guide discusses proper application of thermal overload devices in valve motors that are used in safety systems. Since the temperatures during a degraded core accident might be higher than those during a LOCA, the guide should be revised to consider the higher temperature when selecting the thermal overload trip setpoint. (1)

1.121 Bases For Plugging Degraded PWR Steam Generator Tubes. (8/76) This guide describes how to decide whether an inspected steam generator tube needs to be plugged. The margin of safety against tube failure under accident conditions should be considered. The guide might be revised to consider the margin of safety under a degraded core situation. (1)

1.124 Service Limits And Loading Combinations For Class 1 Linear-Type Component Supports. (1/78) This guide provides design guidelines for Class 1 linear-type component supports. If the loading on the support during a degraded core accident is more severe than that during a design basis LOCA, then the guide might be revised to consider the more severe loading as a design basis. (1)

1.130 Service Limits And Loading Combinations For Class 1 Plate-And-Shell-Type Component Supports. (10/78) This guide presents design guidelines for designing

Class 1 plate-and-shell-type components supports. If the loading on the support during a degraded core accident is more severe than that using a design basis LOCA, then the guide might be revised to consider the more severe loading as a design basis. (1)

1.131 Qualification Tests Of Electric Cables And Field Splices For Light-Water-Cooled Nuclear Power Plants. (8/78) This guide describes qualification tests for safety-related electrical splices. Since a degraded core might impose more severe environmental conditions on splices in-containment than a design basis LOCA, the guide might be revised to use the more severe environment for qualification tests. (1)

1.139 Guidance For Residual Heat Removal. (5/78) This guide presents requirements for the residual heat removal system, which must provide long-term decay heat removal for a shutdown reactor. If the RHR system is to be used for long-term cooling following a degraded core accident, then this guide might be revised to include appropriate degraded core requirements, such as filters and shielding for the RHR system to handle the higher contamination levels of the primary coolant. (2)

1.145 Atmospheric Dispersion Models For Potential Accident Consequence Assessments At Nuclear Power Plants. (8/79) This guide describes acceptable models for calculating atmospheric diffusion of radioactive releases. If different forms or isotopic mixes of material are dispersed as a result of a degraded core accident, then the dispersion models may need modifications.

5.0 ADDITIONAL REGULATORY IMPACTS

Several other selected portions of the regulations were briefly reviewed for impact of degraded core accident considerations. They are not specifically addressed in Section 3 of this report. These sections include 10 CFR Parts 19, 21, 51, 71, 73, 140, 170 and 55 (briefly discussed in Section 3). Of these, only Parts 140 and 170 manifestly require more detailed review. Part 140 could require determination of whether indemnification requirements must be made more substantial if accidents with more severe potential consequences are considered. Part 170 should be reconsidered to determine the impact of degraded core accident review on the fee structure for reactor licensing.

Several of the other Parts cited address areas where the regulated activity and other regulatory documents could be affected. In these cases, however, there is sufficiently little detail contained within the regulations such that no conflict arises with the addition of degraded core considerations. For example, operator licensing (addressed in Part 55) must clearly reflect the procedures to be followed in the event of degraded core accidents, and the use of any modified equipment resulting from requirements for degraded core accident mitigation. As written, however, 10 CFR Part 55 appears sufficiently general to cover such changes. Note that this means that changes to these regulations would not be necessary, but does not imply that additional specification to these parts would not be desirable if degraded core accident requirements were instituted. Changes to more clearly delineate the new scope of licensee responsibility in several of the parts of the CFR cited could be beneficial. Furthermore, changes in NRC Regulatory Guides, industry standards, the Standard Review Plan, and other documents more detailed than the regulations would be likely in the areas associated with these parts of 10 CFR.

In addition to the above regulatory review, certain Division 5 and 7 Regulatory Guides were reviewed due to the anticipated problems which have been identified regarding such concerns as fuel removal and transport, and criticality estimates associated with the TMI accident. This review concluded that these regulatory guides appear to be sufficiently broad in scope so as to not directly conflict with degraded core accident considerations.

However, it is recognized that additional difficulties may exist in the handling of damaged fuel. Therefore, licensees may need to develop other procedures and techniques for handling damaged fuel that meet the current regulations. If not, special cases in the current requirements or guidance may be indicated.

For example, it is noted that Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks," could require revision to reflect greater internal stresses due to increased gas release from the damaged fuel. It is likely, however, that such considerations may be determined on an ad hoc basis as required.

6.0 SUMMARY

The foregoing sections have described possible changes to certain NRC regulations and regulatory guides as a result of including degraded core accidents in the licensing process. Those sections strive to place these findings in their proper perspective, that is, that they are preliminary in nature and should be viewed as a "first cut" assessment of the magnitude of the modification that might be required and as illuminators of the problems that must eventually be resolved when and if the regulatory base is expanded to include degraded core accidents.

The following highlights represent the significant issues identified during this work:

- Possible regulatory changes have been identified during this study with the aid of particular assumptions and a structure of required functions and accident conditions. However, detailed identification of required regulatory changes has been significantly limited by (1) lack of a precise definition of a degraded core accident, both in terms of sequence of events leading to core damage and in the state of fuel damage as a result of the accident, (2) the uncertainty as to how degraded core accidents are to be included in the regulatory base, and (3) uncertainty as to whether existing requirements are sufficiently conservative to cover degraded core accidents.

- Those regulations and guides which appear most likely to require change are those that relate to the current assumptions, analyses, etc., for LOCA events, since the regulations more explicitly address the design bases for these events. This suggests that consideration of degraded core accidents should occur within a framework that addresses the overall design bases and safety analyses for nuclear power plants. Other segments of the regulatory base outside the scope of this study, such as the Standard Review Plan and industry standards and codes, may also require change. The nature of these documents may require more specific definition of the changes to be made in these documents than those changes identified in this study. Such changes cannot be considered profitably without the definition described in the following item.

- A complete, detailed definition of the degraded core accident, including event sequences, fuel state, and regulatory approach, must precede any meaningful continuation or extension of this or similar efforts. This definition must include information useful to the designer. The nature of such information may be surmized from the structure presented in this report. It would include, for instance, limits, rates of change, and durations of environmental conditions cited in Section 2.4.

- There are numerous modifications to the regulatory base underway at NRC, some of which bear directly on the issue of degraded core accidents. It was outside the scope of this effort to review the current state of these developments to assess the impact of degraded core accidents on them.

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