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STAFF EVALUATION OF THE INDUSTRY DEGRADED CORE RULEMAKING (IDCOR) INDIVIDUAL PLANT EVALUATION METHODOLOGY (IPEM) FOR PRESSURIZED WATER REACTORS

EVALUATION OF THE
IDCOR IPEM FOR PWR

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Industry Degraded Rulemaking Group (IDCOR)
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1. INTRODUCTION

On August 8, 1985, the U.S. Nuclear Regulatory Commission issued a policy statement on severe accidents (50 FR 32138). On the basis of the available information at that time, the Commission concluded that existing plants pose no undue risk to the public, and the Commission sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. Thus the Commission withdrew the advance notice of proposed rulemaking on Severe Accident Design Criteria published on October 2, 1980. However, the Commission emphasized that systematic examinations of existing plants are needed.

For existing nuclear power plants, the Commission specified the formulation of a systematic approach to an examination of each plant now operating or under construction for possible vulnerabilities to severe accidents. These individual plant examinations (IPEs) are intended to identify the plant-specific vulnerabilities that contribute significantly to the overall risk from severe accidents. NRC and industry experience with plant-specific probabilistic risk assessments (PRAs) indicate that systematic examinations exposed relatively unique vulnerabilities to severe accidents. Experience also showed that the plants' unique risks could be reduced by low-cost improvements.

Through an initiative by the Industry Degraded Core Rulemaking Group (IDCOR), two separate methodologies were developed for evaluating generic applicability of the reference plant results to other individual plants. IDCOR submitted the methodologies to provide a framework for performing the systematic examinations required by the severe accident policy statement. The methodologies, one for BWRs and one for PWRs, were structured to examine the plant's design and operation to ascertain if there are any vulnerabilities from the standpoint of severe accident prevention or mitigation. The vulnerabilities to core damage are addressed in the system analysis or front-end portion of the IDCOR IPEM. The ability to mitigate the consequences of a damaged core is examined in the source term analysis or back-end portion.

In May 1986, IDCOR submitted to the NRC staff the first package of documents describing the methodologies. The methodology, termed the Individual Plant Evaluation Methodology (IPEM) was developed as part of IDCOR Task 85.3, "Generic Applicability Report," and are documented in IDCOR Technical Report T85.3. The T85.3 report is composed of four volumes -- the A1 and A2 reports covering the system analyses and source term analyses for PWRs and the B1 and B2 reports covering analogous material for BWRs. Subsequent revisions to the methods were submitted in December 1986 (Refs. 1 through 5) and March 1987 (Refs. 6, 7, and 8).

A preliminary review of the 1985 version of the IPEM was performed by the staff and its contractor, Brookhaven National Laboratory. A resulting set of comments regarding the methodology was provided to IDCOR by a letter dated September 9, 1986 (Ref. 9). IDCOR responded to these comments by a letter dated December 10, 1986 (Ref. 10). The staff's evaluation was based on review of the December 10, 1986 responses to staff concerns; the revised T85.3 documents; comments received from other parties, e.g., Sandia National Laboratories (Ref. 11); IDCOR responses (Ref. 12) provided independently to the ACRS; and documented test applications of the IDCOR IPEM for seven plants (Refs. 13 through 19). In addition, the staff's evaluation reflects the insights obtained by the staff and its consultants through (1) meetings with IDCOR and utilities at four different plants for which test applications were performed and (2) participation by the staff and its consultants in a walk-through exercise at one plant.

The staff has completed its review of the IDCOR IPEMs. The results of the staff's review are provided in two separate evaluation reports, one on the PWR IPEM and the other on the BWR IPEM. This report provides the staff's evaluation of the IDCOR PWR IPEM.

As a result of the staff's evaluation, a set of enhancements has been identified for the performance of an effective and useful IPE, including identification of potential area of improvements as called for in the severe accident policy. These enhancements are summarized below and discussed in more detail in the later part of the report.

A. <u>Front-End Enhancements</u>

(1) Only a small number of support systems end state were judged to be important and included in the IPEM. The staff recommends that most of the support systems traditionally used in PRAs in the analysis be included since potential plant-specific support systems end-state vulnerabilities can be overlooked. A list of support systems recommended for inclusion in the IPEs is provided in Section 2.1.1.

(2) Symmetry is assumed between support systems and front-line systems in the IPEM. For asymmetric cases this can lead to incomplete end-states and over or under estimation of accident sequence frequencies. Therefore, symmetries should not be credited unless the support system configuration is confirmed by the IPE team to possess this property.

(3) Misalignment of shared systems in multiple unit plants must be considered in the IPEM.

(4) Underlying causes of vulnerabilities among the screened sequences are not rigorously identified by the IPEM. Sequences must be further expanded to identify specific components, plant conditions

or behaviors, common cause failures, or human actions that dominate plant outliers. This expansion is also necessary to objectively identify any potential fixes.

(5) Subtle dependencies among plant systems with regard to certain accident conditions are not adequately addressed. Development of a more detailed (Appendix C to NUREG/CR-2815) (Ref. 20) and comprehensive dependency matrix (or equivalent) is required.

(6) Treatment of common cause failures in the IPEM is inadequate. Use of the qualitative and quantitative methods detailed in NUREG/CR-4780 (Ref. 21) (or equivalent) is required.

(7) Use of sensitivity studies to determine the more vital assumptions is required.

(8) The use of failure data from PSA Procedures Guide (NUREG/CR-2815) (Ref. 20) is required.

(9) Better treatment of human recovery actions is required. NUREG/CR-4834 (Ref. 22) provides adequate guidance for such treatments.

(10) Certain sequences of events make the front-end and back-end analyses dependent. This must be appropriately treated in the IPE.

B. Back-End

As a result of the staff review of the IDCOR IPEM back-end analysis, we conclude that it is unacceptable for performing the IPE since it does not account for uncertainties and precludes several phenomena and alternative issue outcomes recognized as plausible by the reactor safety community. Appendix 1 to the IPE generic letter provides guidance for evaluating containment performance.

2. STAFF EVALUATION

In January 1985, IDCOR began the development of methodologies for evaluating the generic applicability of their reference plant analyses and results to other individual plants. The IDCOR PWR IPEM systematically searches to identify those plants that are vulnerable to severe accidents and as such could pose an undue risk. Since no precedent exists for the regulatory acceptability of severe accident methods for consistency with the severe accident policy, the IDCOR PWR IPEM was evaluated against the following items.

1. The capability to discover severe accident vulnerabilities and potential areas of improvement.
2. The degree to which the methods provide for a systematic examination of the plant.
3. Identification of the limitations of the scope and results.
4. The capability to assess the effects of proposed fixes for resolution of USI A-45.

The results of the staff's evaluation of the IDCOR PWR methodology is provided in four parts. These correspond to (1) the system (front-end) analysis, (2) the source term (back-end) analysis, (3) the front-end to back-end interfaces, and (4) the study results.

2.1 System (Front-End) Analysis

The IDCOR IPEM makes direct use of past PRA experiences, it requires a small fraction of the resources required to perform a full-scope PRA, and it represents an effective tool for PRA technology transfer, especially for those utilities with very limited or with no PRA experience.

The IPEM features are:

1. The IDCOR IPEM (with the staff's enhancements) is estimated by the staff to require a level-of-effort commitment equivalent to Level-1 PRA.

2. The IDCOR IPEM involves the direct use of previous PRA experiences and insights to sharpen the focus of the analytical effort and to aid in identifying plant outlier features. It takes advantage of system similarities among plants and provides guidance to allow for plant-specific design and configurational and operational differences to be incorporated into the analysis.

3. Systems notebooks include a substantial amount of PRA-type data (e.g., system success criteria and support system dependencies).

4. The IDCOR IPEM takes advantage of insights derived from the past PRA studies and their review, as well as deterministic safety evaluation. It emphasizes what has been identified as risk dominating initiators and incorporates such events as interfacing system LOCAs and transient-induced LOCAs. The IDCOR IPEM explicitly models support systems, dependent failures, impaired containment, and human factors.

5. The IDCOR IPEM system analysis method guides utility analysts through the investigation of most plant features that previous PRAs and operating experiences have identified as specific vulnerabilities to severe accidents. It concentrates on those areas where system vulnerabilities, key operator actions, essential front-line system and support system functions have been identified in previous studies of similar plants to be important.

6. The use of the support state concept and support system event tree separates the effect of the initiating events on support systems from local failure of the support system. This simplifies the analyses necessary to generate, manipulate, and quantify plant specific accident sequences. It also allows efficient performance of sensitivity analysis and reevaluation of those proposed modifications that will not change plant support.

The IDCOR IPEM relies heavily on the analyst's judgment. Enhancements to the methodology for the performance of an IPE are discussed below. These are divided into the following five areas: (1) system modeling/fault tree analysis, (2) analysis of dependent failures, (3) operator actions and human reliability analysis, (4) data base, and (5) applicability to Babcock & Wilcox and Combustion Engineering plants.

While the depth of the IDCOR IPEM analyses does not provide assurance that all risk significant features that are possibly unique to a plant will be captured in the IPE, we believe a majority of such risk significant features will be identified provided the IDCOR IPEM guidance, the recommended level of effort, and the staff's enhancements are implemented.

2.1.1 System Modeling/Fault Tree Analysis

The IDCOR IPEM uses the concept of support states to separate the influence of initiating events from local faults of support systems. This concept minimizes the amount of cutset manipulation necessary for the generation and quantification of accident sequences. While this approach is conceptually adequate for generating accident sequences, the analyst must be careful to generate only independent and unique support states. The IDCOR IPEM has not provided adequate guidance for the creation of such unique support system end-states. For example:

1. Only some of the support systems that are prejudged to be critical to front-line systems are considered for inclusion into the support states. Other support systems can also be critical in certain plants. Inclusion of most of the support systems is recommended. The following support systems have been shown to be important and as a minimum should be included in the analyses: (a) electric power system (AC and DC), (b) ESF actuation system, (c) instrument air system, (d) heating ventillation and air conditioning (HVAC) system, (e) service water system and (f) component cooling water system. However, the IPE team is encouraged to include other support systems where appropriate.

2. In order to minimize the number of independent and unique support states, the IDCOR IPEM takes advantage of highly symetric support system configuration (i.e., when train 1 of support system(s) only supports train 1 of the system(s) being supported). This way, the support states can be described at the support system level and not the train or segment level. However, most of the PWR plants exhibit a small degree of asymmetry in their design and configuration. As a result, certain highly contributing support states can be overlooked, which in turn impacts the accident sequence evaluation and vulnerability determination. The asymmetry problem may also exist within a support system. In this case the asymmetric parts of a support system must be explicitly modeled in the support system event tree that is used to generate the support states. Division of the support systems to train or segment levels ensures that all independent and unique support states are calculated. This division would obviously increase the number of support states, but incorporates more rigor into the IPE.

3. The IDCOR IPEM allows credit for support systems that can be shared by multiple units. It is not clear from the guidance that misalignment of these shared systems will be adequately accounted for in the unique support states. Possible misaligned configurations of support systems must also be explicitly modeled and accounted for in the calculation of the support states (e.g., diesel generators that are shared between two units).

4. There is some ambiguity in the guidance in the system analysis methodology regarding the separation of front-line systems and support systems. For example, the safeguard actuation is categorized as a support system (Table 2.2-7), yet it also appears as a top event in LOCA trees (Figures A.1-1, A.3-1, and A.4-1). The IPE team should establish explicit and consistent criteria for treating top and basic events.

The IDCOR PWR IPEM suggests the use of a modular fault tree concept. The fault trees are developed, or trees from a reference plant are modified, at independent segment or train levels in order to make numerical calculations simple. The end result of the fault tree analysis is the evaluation of a point estimate for the top event. When the fault tree numerical results are combined with the front-line event trees along with the support states, dominant accident sequences are generated and their frequencies are calculated. Therefore, the end result is a frequency for each accident sequence. One can examine the dominant sequences to conclude major contributing front-line systems or support systems (or vulnerabilities), but this examination cannot reveal the basic source of such vulnerabilities. In order to reveal the basic source of vulnerabilities, one must know the major contributing components. This would not only allow systematic identification of basic contributing sources of vulnerabilities, but can aid in searching for potential fixes. For the purpose of identification and objective evaluation of vulnerabilities, applicable dominant accident sequences will need to be further expanded, through a Boolean expansion, to generate dominant contributing components (e.g., specific plant components, human actions, common cause failures, and system dependencies).

Since the IDCOR IPEM heavily relies on the reference plant results and analyst's judgment, the accident sequences generated through system modeling/fault tree analysis are approximate. No provisions for the performance of a formal sensitivity or importance calculation are made to highlight, and thus better treat, sensitive or probabilistically important elements of the plant-specific accident sequences. Use of sensitivity and/or importance analysis is required. PSA Procedure Guide (NUREG/CR-2815) (Ref. 20) discusses these methods.

2.1.2 Analysis of Dependent Failures

One of the primary benefits of performing systematic examinations has been to identify dependencies within a system or between systems. While a detailed support state concept and system fault trees would incorporate a majority of these dependencies into the IPEM model, subtle dependencies can still be overlooked. For example, one of the NUREG-1150 (Ref. 23) insights from the Sequoyah core damage sequences has been that use of sprays for containment pressure suppression may rapidly deplete the RWST inventory and may, as a result, cause high-pressure injection failure if recirculation fails. Since most of these dependencies are sequence dependent, they must be treated following the generation of accident sequences through a careful examination of each sequence. Support systems of U.S. nuclear plants vary widely from plant to plant even in plants that are of a similar class and have the same set of front-line systems. Therefore, it is important to ensure a complete dependency analysis and to document the results in a scrutable form that would include the failure modes and timing.

The IDCOR IPEM guidance for the treatment of common cause failures is not sufficiently rigorous to identify plant-specific vulnerabilities originating from common cause failure. It is important to note that the source of a large portion of plant vulnerabilities found in the past has been common cause failures. Therefore, it is important to carry out a detailed and scrutable method of screening accident sequences for potential contribution from common cause failures. In order to qualitatively incorporate more rigor into the analysis, one can expand the screened sequences using Boolean techniques and explicitly model common cause events in the relevant fault trees. Major contributing plant vulnerabilities originating from common cause failures must then be carefully examined so as to reveal possible root causes of such failures and determine likely fixes.

Use of one of several methods suggested in NUREG/CR-4780 (Ref. 21) (e.g., one of the parametric methods such as the beta factor) is recommended. It is our view that insights on the relevance of this type of dependent failure can be enhanced by using sensitivity analyses in conjunction with parametric models and that such studies should therefore be performed by utilities using the IDCOR IPEM. Guidance for performing the sensitivity studies is included in the PSA Procedures Guide (NUREG/CR-2815, Section 6.4) (ref. 20).

2.1.3 Operator Actions and Human Reliability Analysis

Proper treatment of recovery actions is necessary for identification of vulnerabilities. In the case of sequence-dependent recovery actions, timing is recognized as the most important parameter on which errors of cognition estimates are based. The IDCOR IPEM recognizes this importance but provides little guidance for estimating the time interval available for recovery. IDCOR IPEM users should provide appropriate justification for their estimates of errors of cognition. Sequences of events in which a recovery action will cause an adverse effect if additional components fail have not been treated. Investigation of this type of human recovery actions is very important when identifying plant-related severe accident vulnerabilities. In addition, in small LOCA event trees, operator actions appear as event tree headings. Depending on the reactor design, there needs to be a specific guidance to treat operator actions either as basic events or as event tree headings.

For human reliability analysis (HRA) considerations, the PWR IDCOR IPEM refers to an outdated version of NUREG/CR-2815. The present version of NUREG/CR-2815 is Revision 1, dated August 1985. The HRA section of NUREG/CR-2815, Rev. 1 (Section 4.3) is significantly different (updated, expanded, and enhanced with 10 HRA analytic methods) from its NUREG/CR-2815 predecessor and should be used instead. Better treatment is given in NUREG/CR-4834.

Human error screening is addressed in NUREG/CR-2815, Rev. 1, Section 6.3 (Ref. 20). The screening process describes how to determine the potential importance of each error to the core damage frequency. This involves a list of potential human errors together with screening values. Human errors that contribute significantly to core damage frequency ("important errors") are then studied further as part of the Human Performance subtask; errors that do not contribute significantly are deemed not worthy of further study. The list of errors and the output list of important errors should be supplied to the NRC by the IPE team along with a detailed statement of the criterion of importance.

Besides the aforementioned screening process, another acceptable screening approach is described as the Systematic Human Action Reliability Procedures (SHARP) (EPRI-NP-3583) (Ref. 24) dated June 1984. SHARP was developed to provide a structured approach to the incorporation of human interactions into PRA. The SHARP screening techniques were formulated to identify and select only the most important/significant human interactions for further analysis. The three SHARP techniques are qualitative judgmental screening, quantitative coarse screening, and quantitative fine screening. These techniques are discussed further in NP-3583, Section 2, which includes a discussion of guidance in selecting a technique to eliminate from consideration human interactions that are not significant to core damage frequency.

The IPE team should use the revised version of NUREG/CR-2815 in performing human reliability analysis rather than the version referenced in the IDCOR system analysis methodology. The team should also determine whether a recovery action can adversely affect mitigation of an accident in light of additional component failures. The team should treat the effect of such human recovery action in the accident sequence quantification process.

2.1.4 Plant Familiarization and Search Exercises

Plant familiarization and search exercises are tools of increasing importance for identification of system interactions, verification of as-built configurations, and validation of procedures as implemented by the plant operators. The IDCOR IPEM recommends approaches to the proper performance of visual inspections including (1) a mix of expertise for the walk-through teams, (2) defined inspection criteria for each hazard considered, and (3) a number of tables to document the team findings. Participation in part of a PWR walk-through led us to conclude that strict adherence to the methodology guidance is needed for a walk-through or a talk-through process. This process should be iterative in nature, starting with plant familiarization and evolving from this point to search for answers to questions raised during the analytical effort. Timing, scope, and mix of expertise should be integrated into the IPE analyses. We realize that specific guidance in this area can be difficult; however, SRP (Ref. 25) Appendix 7.B provides a general agenda that could be used for the development of a plant-specific agenda for the plant familiarization. The IPE team should give this process a prominence worthy of its importance, especially if the IDCOR IPEM scope is extended in the future to consider external events beyond internal floods.

2.1.5 Data Base

The IDCOR IPEM recommends the use of the Interim Reliability Evaluation Program (IREP) (NUREG/CR-2728) (Ref. 26) generic data base. However, some of the IPE applications used Zion or Oconee data as generic data. A unified and up-to- date generic data base would provide a consistent and uniform basis for quantification. Updates of the data base included in the PSA Procedure Guide (NUREG/CR-2815, Appendix C) provides a better data base.

2.1.6 Applicability to Babcock and Wilcox and Combustion Engineering Plants

The IDCOR PWR IPEM was based primarily on a Westinghouse reactor design and is structured to assess the applicability of the IDCOR reference plant results to other PWRs. However, various levels of additional information and modifications are expected in an application to Babcock and Wilcox (B&W) and Combustion Engineering (C-E) plants as demonstrated by the test application performed for the Oconee plant (B&W reactor design) using an earlier version of the IDCOR IPEM. For example, B&W plants show a unique response to those anticipated transients involving overcooling and undercooling events as well as small LOCAs. This is mainly due to the small heat sink represented by the once-through steam generator, which is a unique design feature of the B&W plants. Accordingly, the operator actions and available times for recovery are significantly limited. These design variations can influence the identification of plant-specific initiators, modeling of accident sequences, system analysis, and data quantification. Therefore, the IPE team should compare the features that are specific to the B&W or C-E plant with those of the Westinghouse reference plant to determine the significance of the features on the fault tree matching process. A list of the potential vulnerabilities previously identified by PRAs for B&W and C-E plants should also be considered by the IPE team. For example, in station blackout sequences of loss of offsite power and the resulting loss of the emergency feedwater motor-driven pumps, if the turbine driven auxiliary feedwater system pump fails, core damage will occur soner for B&W plants. This is due to the smaller B&W plant inventory which results in a short time available to the operator to recover offsite power prior to onset of fuel failure. In non-B&W plants, because of larger inventory than B&W plants, longer time is available for the operator to recover offsite power. If recovery of offsite power occurs within about two hours, the motor-driven auxiliary feed pump can then keep non B&W plant cooled.

Some specific examples at each stage of the IPEM follow:

1. Initiating Event Identification. The integrated control system (ICS) is a control system unique to a B&W plant. It provides fast control of steam flow and pressure, feedwater flow, and reactor power by manipulating various control devices. Failures in the ICS or in the ICS power supplies are possible candidates for B&W plant-specific initiating events. Also, failures in the ICS during plant transients may cause plant operators to misdiagnose the symptoms so that the human error rate might be high.

2. System Analysis and Event Tree Development. B&W, C-E and Westinghouse plants are quite different in design and operation, which makes the respective systems analysis and event tree different. For example, the large LOCA event tree given in Figure A.1-1 of the PWR system analysis methodology does not necessarily represent the accident progression in B&W or C-E plants. High-pressure injection may not deliver enough cooling or makeup of reactor coolant system inventory if the same success criterion as that used for small LOCA is used. Separate event trees and success criteria for the different reactor designs appear to be needed. Some B&W plants use the low-pressure recirculation pumps to solve NPSH problems of high-pressure recirculation pumps. Therefore, the loss of low-pressure injection pumps results in the loss of the high-pressure recirculation function. Some C-E plants, unlike Westinghouse and B&W plants use a high-pressure system whose pumps can perform both

high-pressure injection and high-pressure recirculation functions. During the injection phase, the system takes suction from the refueling borated water tank, while during the recirculation phase the system will be realigned to the containment sump. Therefore, the IPE team should provide explicit success criteria for each B&W or C-E plant.

2.2 Source Term (Back-End) Analysis

IDCOR stated that the purpose of the IDCOR IPEM source term analyses is:

> "...the purpose is to determine if the potential for fission product retention in a given plant is comparable to or better than that determined for the most similar IDCOR reference plant. For this reason the simplified source term methodology is directed toward estimating the environmental releases for the dominant severe accident conditions and determining whether there is a difference in plant design which would substantially increase the releases over those of the reference plant."

On this basis, IDCOR contends that it is "appropriate and sufficient" that the methodology focus on station blackout sequences. Such sequences involve loss of all containment heat removal capability and a dried-out (water-depleted) debris bed configuration. They lead to containment failure, and a great deal of the proposed IPEM quantification deals with the approximate estimation of the resulting releases to the environment.

Briefly summarized, the PWR source term methodology consists of a simplified containment event tree (CET) and an approach for assigning a source term to each CET end state. Source terms are assigned in the methodology as indicated in Table 1.

Table 1 - PWR CET and Source Term Releases

IS CONTAINMENT NOT BYPASSED	IS CONTAINMENT ISOLATED?	IS THE DEBRIS COVERED AND IS CONTAINMENT HEAT REMOVAL AVAILABLE?	END STATE/RELEASE QUANTIFICATION
Yes	Yes	Yes	1. Insignificant
Yes	Yes	No	2. Determine using calculational scheme provided
Yes	No	Yes	3. Much less then PWR-2, e.g., noble gases plus 5% volatiles
Yes	No	No	4. Perform detailed analysis if required by probability
No	--	--	5. Assess using checklist provided

The CET is entered once for each plant damage state identified in the front-end analysis. In applying the CET within the context of the IPEM, rather than quantifying each branch of the CET, the analyst assigns an affirmative or negative response to the CET top event questions based on information developed as part of the front-end analysis. Only sequences with releases corresponding to CET end-states 4 and 5 are flagged in the methodology for further investigation. In addition, the methodology identifies several plant features that may invalidate the simplified approach for estimating source terms and require analysis outside the methodology. For PWRs with ice condenser containments, the source terms are directly assigned for CET end-state 2 based on the results of previous IDCOR calculations.

In the sections that follow, the results of the staff's evaluation of the IDCOR PWR source term methodology are provided. It should be noted that the documented test applications of the IDCOR IPEM played only a minor role in this evaluation since the test applications either did not involve the use of all portions of the source term methodology or were not adequately documented.

2.2.1 Containment Event Tree Structure

The PWR source term methodology consists of a simplified CET with three top event questions. These address (1) whether the containment is bypassed by the initiating event, (2) whether containment isolation has succeeded, and (3) whether the debris bed in the reactor cavity is covered and containment heat removal is available. The simplicity of the CET derives in part from the fact that it was developed based on insights obtained using the Modular Accident Analysis Program (MAAP). Many of the assumptions in the MAAP code received some staff review over the past several years as part of the NRC/IDCOR Technical Exchange meetings. Key differences between many of the NRC and the IDCOR models and assumptions were identified during the meetings and were consolidated into 18 NRC/IDCOR Technical Issues. On the basis of IDCOR analyses and modeling changes described in IDCOR Program Technical Report 85.2, (Ref. 27), the issues were disposed of by the NRC either (1) by determining that the differences had little effect on the results or (2) by developing interim positions that conservatively treated the phenomena. In References 28, 29, and 30, the staff positions on these phenomenological issues are presented. The staff finds that the models and assumptions in MAAP and the lack of treatment of issue uncertainties in the code preclude several phenomena and alternative issue outcomes recognized as plausible by the reactor safety community and, therefore, the sorce term methodology is not adequate for meeting the IPE objectives in Section 2.2.2.

While the issues addressed by the top event questions in the CET are indeed risk important, the lack of treatment in the CET of several other aspects of severe accident releases that have generally been considered in most PRAs and found to be important is considered by the staff to be a major limitation of the source term methodology. Foremost, the CET does not consider the potential for containment failure due to certain ex-vessel phenomena, namely, steam spike (resulting from rapid core quench), direct containment heating, and hydrogen combustion. As a result, the CET does not recognize the potential for early containment failures; except for a limited number of situations that the methodology specifies as requiring additional analyses, all containment failures are treated as late failures. Also, in formulating the CET, an optimistic characterization of certain phenomena or issue outcomes was chosen without adequate basis. In broad terms, the staff finds the IDCOR source term methodology to be unacceptable. The staff's concerns may be summarized as follows:

1. The IDCOR source term IPEM is too narrowly focused,
2. The IDCOR source term IPEM does not adequately reflect remaining generic issues and associated uncertainties, and
3. Containment systems performance is not adequately integrated with containment phenomenology.

Details on these areas of concern are given in the following sections.

2.2.2 Approach and Objectives

The staff recognizes that the IDCOR IPEM needs to provide a mechanism for seeking out plant-specific differences that would substantially increase the releases over those estimated for the dominant severe accidents in the reference plant (i.e., in IDCOR's terminology, identifying "outliers"). But it is the staff's opinion that the IPE objectives should also include:

1. Appreciation of severe accident behavior, and
2. Recognition of the role of the mitigation systems and accident management.

It is the staff's belief that such objectives can be attained to a reasonable degree with relatively modest commitments of effort, by the utilities, and that the IPE provides a unique opportunity for doing so. This conclusion is based on existing detailed calculations which:

1. Have already mapped a rather broad range of containment failure timing and containment atmosphere conditions into a set of release categories and consequences,
2. Have demonstrated, that containment loads depend on a few key phenomenological behaviors, and
3. Have concluded that the system responses and human actions are of decisive importance.

From these observations, the following deductions are provided for the IPE overall approach.

1. The approach could beneficially focus on containment failure mechanisms and timing. Releases should be based on corresponding release categories and associated detailed quantification from reference plant analyses.
2. All classes of sequences with significant probability (front-end results) should be considered.

3. System/human response should be integrated probabilistically with phenomenological aspects into simplified but realistic containment event trees. Allowance should be made for recovery or other accident management procedures (particularly for long-term responses).

On this basis, the back-end analysis also emerges as system oriented. The key thrust of the suggestion here is that the source term quantification should be judiciously integrated into the containment phenomenology to address containment response as well as failure modes and timing. The fundamental premise is that such an enhanced scope would involve the utility to an appropriate level necessary to understand the possible range of severe accident behavior in their plants and thus be better prepared to mitigate severe accident progression and consequences.

2.2.3 Generic Issues

Starting with the ZIP study (Zion/Indian Point PRAs and associated NRC studies, NUREG-0850) (Ref. 31), our understanding of severe accident phenomena has developed rapidly in the past few years. The Containment Loads Working Group (CLWG) effort (NUREG-1079) (Ref. 32), the IDCOR program, and the Severe Accident Risk Research Program (SARP) have been key contributors to this development. With this improved understanding came a gradual convergence of the NRC and industry's views in many areas of quantifying severe accident phenomena and associated containment loadings or source terms. Still, a number of issues are difficult to quantify, and there is considerable uncertainty in the results. In a recent series of NRC-IDCOR meetings, these issues were dealt with in depth and the nature of the difficulties was distilled and clarified (Refs. 28, 29, and 30). The NUREG-1150 (Ref. 23) results, when considered against the IDCOR reference plan analyses (including their treatment of uncertainties), provide a further illustration of the origin of these difficulties and their risk implications. Clearly, a number of risk-significant phenomenological aspects of severe accident behavior have so far defied reliable quantification.

The vast majority of these issues is associated with the so-called high-pressure sequences. Such sequences involve core meltdown and vessel meltthrough from a high primary system pressure condition. Station blackout provides a typical example.

As a consequence of the high primary system pressure, the in-vessel portion of the accident involves strong natural circulation flows (Ref. 33) between the core, the upper internals, and steam generators. The natural circulation flows provide energy redistribution of such magnitude that it may induce primary boundary failure before core slump. The concern is that such failure could occur in the steam generator tubes yielding containment bypass. The fission product transport involves augmenting the velocity of many of the less volatile radionuclides (UO_2, BaO, Mo) in the presence of high-pressure steam. Their transport to other parts of the system (including steam generator tubes) would result in additional heat load and hence enhanced failure potential. On the other hand, gas-phase mass-transport processes at high pressure are found to sharply limit the release of the more volatile fission products (Cs, I, and Te), which would then enhance the source term following blowdown. None of these issues are addressed in the IDCOR IPEM.

For the ex-vessel portion of the sequences, melt release at high pressure implies high-velocity steam flows in the reactor cavity and the potential for large-scale melt dispersal throughout the containment volume. Such dispersal could lead to direct heating of the containment atmosphere and associated pressurization. Clearly, the extent of dispersal would be geometry dependent, but the appropriate manner to quantify such effects is not clear at this time. Furthermore, such a sequence would also give rise to highly dynamic hydrogen release (from that already released during the core heatup/slumping phases and that released during the blowdown/dispersal process) and mixing phenomena. Predicting the potential for the formation of combustible or detonable mixtures or assessing igniter performance is inherently more uncertain under such conditions. Again, containment geometry could play a significant role here, but at this time this role remains unspecified. An attempt to address the dispersal issue (in terms of entrainment and deentrainment) was made in the IDCOR IPEM. The staff and its consultant reviewed the IDCOR screening criteria for direct containment heating and identified several areas where the proposed approach does not provide adequate basis and, therefore, is not acceptable.

To summarize, it is the staff's view that, although entrainment and deentrainment can be relegated to some threshold between inertia and buoyancy forces (i.e., the Kutateladze number), the use of the existing IDCOR threshold is problematic and further confused by the choice of vessel failure area, cavity gas temperature, possible degassing and splashing effects, and predominant particle sizes. In particular, for cases where the cavity is strongly overdriven (i.e., Kutateladze number being much higher than the entrainment threshold) particle sizes could be well below the capillary length, which could affect the deentrainment behavior in the steam generator compartment in yet unknown ways. It would therefore appear difficult to reliably sort out at this time the geometric features of importance and their quantitative effect on dispersal.

It is the staff's view that the CET failure to account for the potential for early containment failure due to ex-vessel phenomena, including steam spike, direct containment heating, hydrogen burn, and the interaction of core debris with steel liner/containment, is unacceptable. It is the staff's position that a low containment failure probability should not be assumed without convincing evidence.

A recent independent review by an NRC panel of experts provided an additional perspective on these issues and made recommendations for their resolution (Ref. 34), namely, "if direct containment heating or containment bypass through steam generator tube failure contribute importantly to risk, this may indicate a need for a hardware modification or a procedural measure to ensure depressurization before primary system failure. An early study of relative merits of the possibilities available would be valuable."

By contrast, a low-pressure scenario presents few remaining areas of uncertainty. They relate to the behavior of deep molten corium pools, coincident steam spike and hydrogen burn, and the long-term behavior of hydrogen (and other combustibles) resulting from deinerting by steam condensing on structures, by the late operation of spray, or by the containment atmosphere coolers. The concerns about deep corium pools arose from experiments with top-flooded melts that exhibited crust formation and long-term isolation of the melt from the water coolant. Such uncoolable configurations would yield continuing concrete attack and a containment loading behavior significantly different from that of coolable ones. The IDCOR IPEM assumes that coolability is equivalent to the presence of water on top of the corium melt. The staff views this as an area of uncertainty and is conducting research to reduce this uncertainty. It is entirely possible that beds below a certain depth (heat loading) will eventually be shown to be coolable, in which case cavity floor area would become the decisive factor in the IPE. It should also be mentioned, however, that gravity spreading to all available floor area becomes increasingly problematic as the magnitude of the area increases. This is because of heat losses (and associated corium melt solidification) and possibly a slurry-like corium state at vessel meltthrough.

With regard to the behavior of hydrogen in the containment atmosphere, Appendix O to the final version of the IDCOR PWR source term methodology (Ref. 8) includes a computation scheme for estimating the containment loading associated with postulated global deflagration of an amount of hydrogen corresponding to oxidation of 75% of the cladding. The methodology, in Appendices A and O, states that if the resulting peak pressure (initial pressure plus pressure rise) exceeds twice the design pressure, a detailed assessment of the containment response should be performed.

In this regard, the staff's concerns are related to the quantities of combustible gases released to the containment from reactor vessel structural materials, local inerting and de-inerting as well as hydrogen mixing and transport. For example, in large dry containments, combustible concentrations build very slowly and only if continuing concrete attack is postulated. Because of the large volume and flow release rates, detonable compositions do not develop unless significant spatial concentration gradients exist. On one hand, a containment atmosphere under such conditions would exhibit strong natural circulation currents that would tend to even out any tendency to stratify. However, on the other hand, a condensation-driven stratification (Ref. 35) mechanism would limit the circulation patterns to compartmentalized structures thus effectively reducing the volume available for mixing.

For the ice condenser, the IDCOR approach is based on the premise that hydrogen from corium-concrete interactions will ignite in the high-temperature cavity atmosphere and burn as it is being produced. Local inerting and air availability (natural circulation loops) on the other hand could alter this conclusion, as a recent study (Ref. 36) seems to indicate.

It is the staff's current judgment that combustible gas behavior should be explicitly considered in the IPEM containment event trees. It is necessary that the IPE team include consideration of gaseous pathways between the cavity and upper containment volume to confirm adequate communication to promote natural circulation and recombination of combustible gases in the reactor cavity.

2.2.4 Systems/Phenomenology Integration

System performance may be integrated into the severe accident phenomenology with help of containment event trees (CETs). On the basis of the rationale provided in Section 2.2.2, we should address all prominent classes of severe accident sequences (as determined from the front-end analysis) and should account for uncertainties. Most importantly, these trees must be mechanistic and allow for accident management actions by providing a realistic road map under all physically meaningful outcomes.

In summary, the staff has reviewed the proposed use of the CET in the IDCOR IPEM, and the effect that this approach would have on the ability of the IDCOR IPEM to identify outlier plant features. We conclude that the use of the CET in a qualitative rather than quantitative manner will not provide utilities the information and perspective necessary to make a judgment regarding the expected level of containment performance for severe accidents, and the frequency of large releases and, therefore, is unacceptable.

To produce the information considered necessary by the staff, the CET needs to be augmented as recommended in the previous sections and then quantified as part of each IPE. The branch point probabilities should be propagated through the CET in such a way that estimates of the likelihood of early containment failure and large releases from containment are developed as part of the IPE.

The staff therefore provided, in Appendix 1 to the IPE generic letter, guidance aimed at providing a quantitative assessment of source term releases and the relative frequency of the various types of releases to be used in performing IPEs.

2.3 Front-End to Back-End Interfaces

The role of interfaces between the front end and back end is particularly important from two perspectives. First, the condition of the system analysis directly influences the capability of the plant to cope with the damaged core. Second, the conditions of some systems designed to preserve containment integrity and control the release of radionuclides also can influence the likelihood of the core becoming damaged. Thus, because the influences can flow in both directions between the system analysis (front end) and the source term analysis (back end), particular attention should be given by the IPE team to these interfaces.

With regard to a staff concern related to front-end and back-end interfaces and the potential for inconsistencies in system availability assumptions, IDCOR has proposed the addition of a sign-off sheet to the methodology that would identify, by sequence, (1) the sequence frequency, (2) whether the containment is bypassed, (3) whether the containment is isolated, (4) the containment system and reactor system availability, and (5) the approximate source term. This sheet would be signed by both the systems analyst and the source term analyst, which would provide added assurance that the availability of key systems is treated consistently in the front-end and back-end analyses. In addition, IDCOR has proposed the addition of an operator interview form to the source term methodology aimed at ensuring that assumed operator actions are given adequate consideration. The staff has reviewed the IDCOR proposal and believes that use of the sign-off sheet and operator interview form, if incorporated into the PWR methodology, will substantially improve treatment of the interfaces. The use of these forms, however, does not replace human reliability analyses for sequences in which operator recovery actions are a key ingredient. The forms also do not address the question of mission times, inventory depletion, and dual usage (e.g., the CST supplies water for vessel injection and containment sprays. Early injection may deplete the water so that it is not available for sprays and vice versa). It is necessary that mission times, inventory depletion, and multiple usage also be carried through the interfaces.

2.4 Study Results

Additional staff views on the IDCOR IPEM and its use are provided in this section. These can be divided into four areas: (1) interpretation of results, (2) documentation of results, (3) study management, and (4) application of results.

2.4.1 Interpretation of Results

The IDCOR IPEM guidance for quantification is not followed up by further guidance about how to systematically review these results for the purpose of deriving insights about the plant design and operations. Instead, it is left to the analyst to evaluate the results and identify the root causes behind dominant sequences or factors driving an outlier feature. Reliance on analyst judgment, despite its drawbacks, may have its best potential during the IPE or right after it has been completed. Following the completion of an IPE, the insights drawn from the IPE and the associated analytical details should be fully documented and retained by the utility in easily accessible form.

Some of the IDCOR IPEM test applications, e.g., for Sequoyah, appear to include a reasonable effort of analysis of IPE results supplemented with importance analyses. Systems, components, operator actions, and support states with major contributions to dominant sequence frequencies were identified as part of this analysis. IDCOR IPEM users should recognize characterization of the dominant accident sequences as an integral part of the analyses.

2.4.2 Documentation of Results

The importance of the documentation requirements within the IDCOR IPEM is twofold. First, the necessary documentation must be available to the IPE team in order to provide meaningful conclusions at the end of its study. Second, the documentation needs to be sufficiently organized to enable an independent review. To achieve these goals, three basic criteria must be satisfied:

1. Explicit documentation requirements should be listed to ensure that sufficient information will be gathered to adequately perform the IPE,

2. Sufficient quality assurance measures should be provided to ensure the accuracy and retention of the documentation packages (notebooks), and

3. Sufficient study management should follow the above actions to ensure that the documentation is indeed gathered and handled as intended by the IPEM.

The IPE should be documented to provide the basis for the findings in a traceable manner. This is viewed as being dealt with most efficiently by a two-tiered approach. The first tier should be the results of the examination that will be reported to and reviewed by the NRC. The second tier is the documentation that must be retained by the utility. The reporting requirement is specified in the IPE generic letter and the review document to be issued shortly.

If the IDCOR IPEM was modified by the utility during an IPE, the utility should also describe and provide the basis for the modifications. The description should clearly identify the differences, the effects on the results, and the basis for the selection of the different method.

2.4.3 Study Management

Insights obtained from previous PRAs involve not only technical matters but also lessons learned in both the management and performance of the PRA itself and the scheduling and execution of technical audits. The IDCOR IPEM reports contain numerous references to the need for PRA project management and technical audits. However, the IDCOR IPEM does not specifically call for independent review and does not provide guidance on formal management. Study management is considered by the staff to be essential since the quality and comprehensiveness of the results coming out of the IDCOR IPEM will depend simultaneously on the rigor with which the utility applies the IDCOR IPEM and the utility commitment to the objectives of the IPE.

The following actions are recommended by the staff as part of the Study Management:

1. Each utility should formally include an independent review to ensure the accuracy of the documentation packages and to validate both the IPE process and its results.

2. Experienced PRA analysts and utility engineers who are familiar with the details of the design, controls, procedures, and system configurations should be involved in the analysis as well as in the technical reviews.

In addition, the following views are offered:

1. A 2 to 3 staff-year effort was recommended by IDCOR as an appropriate level for performing an IPE. The staff has observed that most of the utilities applying the IDCOR IPEM performed their IPEM tests in about 2 to 3 staff- years. In most cases, however the utilities used their existing PRA studies as a source of information. Thus these applications did not involve the use of all portions of the IDCOR IPEM and are considered by the staff to be incomplete tests of the IDCOR IPEM. Also, for those utilities with B&W or C-E plants, modification of the sample fault trees, event trees, and accident initiators provided in the system analysis methodology will require additional effort and a correspondingly higher level of PRA expertise. We estimate that a 4 to 5 staff-year level of effort may be needed for the analysis, documentation, and independent technical reviews before submittal to the NRC. Additional staff level of effort is necessary to upgrade the IDCOR IPEM to Level 1 PRA so that it could achieve one of the goals set forth in the IPE generic letter, namely, establishment of an accident management plan.

2. It is the staff's view that, in such a complex undertaking as an IPE study, computer analysis is essential, especially in evaluating logic trees with a large number of gates or in cases where logic reduction is required to generate minimal cutset information. Thus, unless the subject plant is nearly identical to the reference plant, the staff does not believe that application of the IDCOR IPEM is feasible without computer analysis.

2.4.4 Application of Results

An important aspect of severe accident prevention and mitigation is human involvement. Early recognition of events, availability of procedures specifying corrective actions, and well-trained operators and emergency teams can have a major influence on the course of events in case of a severe accident. An accident management strategy that has the capability to accomplish these functions for each dominant accident sequence despite the degraded state of the plant should be developed. Additional discussion is provided in the IPE generic letter.

3. CONCLUSIONS

The IDCOR IPEM makes direct use of past PRA analyses and operating experiences at a fraction of the resources normally needed to perform a full-scope PRA. The method provides a readily usable tool for the transfer of PRA technology to utilities with little experience in integrated probabilistic assessments. The IDCOR IPEM is not an exact algorithm and the quality and comprehensiveness of the results coming out of the IDCOR IPEM will depend on the rigor with which the utility applies the IDCOR IPEM and on the utility's commitment to the intent of the IPE.

The IDCOR system analysis methodology includes the preparation of System Notebooks that include a substantial amount of PRA-type information (including system success criteria and support system dependencies) that has been useful in probabilistic safety assessments. The IDCOR IPEM particularly emphasizes the systematic examination of support systems since past experience has underscored the importance of the support functions.

The staff evaluated the IDCOR IPEM for use only in the performance of an IPE. Subject to the incorporation of the enhancements discussed in Sections 2.1 and 2.4 of this report, the IDCOR IPEM is considered to be adequate for the performance of an IPE. The specific items identified in this evaluation clarify how the IDCOR IPEM should be enhanced. The potential exists for an IPE team to develop improvements beyond those given in this evaluation. Alternative enhancements may be used by the IPE teams provided their bases are fully described in the IPE report. The staff-identified enhancements are summarized in three groups: general, system analyses, and source term analyses.

3.1 General

Considering the importance of the IPE and the flexibility in PRA analyses, utilities using the IDCOR IPEM should establish an independent review of the technical accuracy of the examination and the validity of the results. The validation process will aid in ensuring that the utility in its primary safety role will assimilate the insights gained from the performance of the IPE.

The documented test applications of the IDCOR IPEM used earlier versions of the methodology. The methodology was subsequently revised without a repetition of the test applications. For example, one important revision concerned the performance of the visual inspections. The revisions were acceptable although the staff has not had the opportunity to observe a test application involving visual inspections that rigorously adhered to the methodology revision. In performing the IPEs, each IPE team should rigorously adhere to the guidance provided in the revised methodology and incorporate the enhancements called for by the staff in Sections 2.1 and 2.4 of this report.

Utilities using the IDCOR IPEM should fully document and retain the insights drawn from the IPE and their associated analytical details. The insights should be supported by appropriate evaluation of the results by importance rankings and sensitivity studies as described in Section 6.4 of NUREG/CR-2815, the PSA Procedures Guide (ref. 20). The IPE results should be documented as discussed in Section 2.4.2.

3.2 System Analyses

Utilities using the IDCOR IPEM should be familiar with the staff added guidance and enhancements in Sections 2.1 and 2.4 along with the supplemental guidance provided on the treatment of common cause failures in Section 6 of the NRC Probabilistic Safety Assessment Procedures Guide (NUREG/CR-2815) and Sections 3, 5, and 6 of the PRA Procedures Guide (NUREG/CR-2300) (Ref. 37).

Utilities using the IDCOR IPEM should clearly justify their estimates of the time intervals available for each recovery action credited in the analysis.

2.3 Source Term Analyses

With regard to the adequacy of the IDCOR source term methodology for use in performing IPEs, we believe that the usefulness of the methodology is limited by its stated objectives as well as in its particulars and is unacceptable.

Regarding the objectives, the staff finds the IDCOR IPEM unnecessarily narrow in focus. It is suggested that the IPE be directed toward the more meaningful goals of developing realistic risk profiles and accident management schemes on a plant-specific basis. To achieve such ends, it would be necessary that each utility develop its own basis for understanding severe accident behavior in its plant (in particular, integrating system/human aspects with the accident phenomenology). This, in turn, requires a quantitative approach to containment event trees and a creative attitude in reflecting on them.

Regarding the particular technical details, the staff finds the IDCOR IPEM position in several key phenomenological issues affecting the high-pressure scenario unacceptable. As a result of the staff's evaluation, we conclude that the IDCOR source term methodology is unacceptable. It is the staff's judgment that the tools are available for proceeding with the IPE. Appendix 1 to the IPE generic letter provides the utility with guidance to proceed with the evaluation of containment performance despite the phenomenological uncertainties.

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ATTACHMENT

Staff's Evaluation of V-Sequence Checklist

A loose interpretation of the V-sequence checklist questions by the utilities could result in a lack of sufficient attention to available means for reducing the frequency and consequences of bypass sequences. The staff in a letter from T. Speis to A. Buhl dated September 9, 1985 requested that IDCOR modify the checklist to include guidance and prescriptive acceptance criteria for each checklist question. Such clarification was considered necessary to ensure that the checklist would not be loosely interpreted by utilities performing the IPE and that undue reliance would not be placed on the auxiliary building as a fission product removal feature. Specific items requiring additional clarification include:

1. The methods and assumptions to be used in defining the low-pressure-system boundary that must be analyzed by each utility.

2. Required actions in the event that only portions of the RHR lines are maintained water filled and acceptable bases (e.g., Technical Specifications/Plant Operating Procedures) for ensuring a water-filled state.

3. The methods and frequencies of hydrostatic testing by which system integrity is demonstrated.

4. Analytical techniques to be used for analysis of piping stresses at elbows and piping supports and the material properties to be used for all stress analyses, e.g., actual material properties with suitable margins to account for uncertainties in modeling, material properties, and construction tolerance.

5. Quantitative criteria regarding the submergence of the failure site required to take credit for pool scrubbing in the auxiliary building and the methods and assumptions to be used to calculate the water additions and the flooding level.

6. The method to be used by each utility for identifying the potential pathways to the environment (including guidance on assessing auxiliary building pressure capability, performance and failure location) and quantitative criteria regarding the minimum release pathway length and intervening structures required to claim applicability of the reference plant analysis.

7. Acceptable bases for ensuring that fire sprays are available and would be actuated (e.g., emergency operating procedures and automatic initiation) and prescriptive criteria regarding the minimum acceptable coverage of auxiliary building (and release pathways) by fire sprays required to claim applicability of the reference plant analysis.

8. Analyses required by each utility in order to claim that ventilation systems will remain intact and effective.