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Add: C. Ryder

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Westinghouse Owners Group

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'87 SEP 24 A10:37

March 13, 1987
52 FR 7950
(23)

OG-87-40

September 22, 1987

Chief
Rules and Procedures Branch
Division of Rules and Records
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Westinghouse Owners Group
Draft "Reactor Risk Reference Document" (NUREG-1150)

The Westinghouse Owner's Group (WOG) has performed a preliminary review of the NRC draft NUREG-1150, "Reactor Risk Reference Document," February 1987.

Due to the extensive volume of material in NUREG-1150 and its supporting documents, the WOG review has been limited. Thus, some significant issues may not have been identified in our review and not included in the general and specific comments attached to this letter.

Based on the technical nature of the comments, we recommend that resolution of these comments as well as those of other nuclear industry representatives be formally addressed prior to final publication of NUREG-1150 and its supporting documents. We also recommend that the NRC and its contractors work with the nuclear industry to resolve the technical concerns in order to better serve the goals of the nuclear industry.

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letter sent 9/29/87

The comments and positions provided in this letter and its attachments have been endorsed by members of the Analysis Subcommittee of the WOG. However, it should not be interpreted as the position of any individual WOG member.

In conclusion, the WOG would welcome the opportunity to work with the NRC and its contractors to makε NUREG-1150 adequately assesses the risk from U.S. Nuclear Reactors.

Very truly yours

Roger A. Newton

Roger A. Newton, Chairman
Westinghouse Owner's Group

RAN/dac

Attachments

cc: WOG Primary Representatives
Analysis Subcommittee
M. Hitchler - WEC 3-22 E
J.L. Little - WEC 4-17 E
N.J. Liparulo - WEC 3-21 E
N. Burns - WEC 3-22 E

WESTINGHOUSE OWNER'S GROUP

POSITION STATEMENT ON

NUREG-1150, REACTOR RISK REFERENCE DOCUMENT

SEPTEMBER, 1987

Westinghouse Owner's Group
Position Statement on
NUREG-1150, Reactor Risk Reference Document

The Westinghouse Owner's Group has conducted a limited review of the NRC draft NUREG-1150, Reactor Risk Reference Document and its supporting documents with emphasis on its impact on Westinghouse plants. While we certainly applaud the level of effort involved in the production of this document, we feel that some aspects of the document and the supporting analyses deserve additional attention. The intent of this "position statement" is to provide some overall comments with regard to NUREG-1150. A supplemental report is provided that includes some detailed comments.

A major objective of NUREG-1150 was to provide a model and assessment of the risk reduction potential for NRC proposed plant design and procedural modifications for use in licensing, inspection and research decision making. The conclusion for all modifications assessed was that "the risk benefits do not clearly outweigh the costs of any of these proposed modifications." This conclusion is consistent with the vast majority of industry and regulatory work to date. Although we strongly endorse this general conclusion, we feel that major biases exist in the areas of plant modification costs, frequencies and definition of core damage sequences and containment phenomenology. These biases severely limit the models effectiveness for use in prioritizing licensing, inspection and research decision making. These biases should be resolved through proper peer review.

One of the other objectives stated in NUREG-1150 was to assess the usefulness of the method in evaluating and providing insights. If NUREG-1150 is to provide insights into plant behavior during severe accidents, shortcuts such as the "SMART" approach should not have been taken. This implies that a thorough analysis was not conducted in order to identify possible outcomes that have not been identified in past PRAs. Modeling shortcuts for data, common cause, human factors, systems interactions and other aspects can be utilized to prioritize efforts and save time. However, the shortcuts should be detailed and reviewed before the continuation of the analyses to prove that the shortcut will not lead to an erroneous conclusion. If the NUREG-1150 is to provide a basis for future regulatory use and to provide a benchmark for future work, the document should provide more detail on the use of this method and sensitivity studies should be conducted to determine the impact of these shortcuts. The SMART approach could lead to erroneous prioritizations that would carry through the entire analyses. Thus, we suggest that the "SMART" approach not be utilized in a document of this magnitude.

Another conclusion of NUREG-1150 is:

> "Large, dry containments have a higher probability of withstanding the effects of severe accidents than do suppression-type containments. For pressure-suppression containments, however, if the pressure -suppression pool or ice bed is available after containment failure, the releases to the environment can be substantially reduced."

The IDCOR program also represented ice condenser and large, dry containments. The IDCOR studies concluded that the PWR ice condenser containments presented approximately the same severe accident risks as PWR large dry containments and that no early containment failures could be expected in either case.

The two sets of conclusions should be reconciled by focusing on the areas that contribute to this divergence. This could lead to satisfactory resolution of issues that have long been disputed between the industry and the NRC. Industry analyses and experimental work must be factored into NUREG-1150 to provide meaningful results.

Furthermore, the use of expert judgment is not described in detail and is poorly documented such that the validity of the judgments can not be determined. If the method used was merely to ask people their opinions and to calculate some central estimate based on these opinions, very little confidence can be placed in the results. Also, experts from industry organizations were not utilized in the expert judgment approach. This appears to significantly bias the results of the expert judgment process, since the industry and the NRC are far apart on issues which dominate the uncertainty. In the final version of NUREG-1150, we recommend that the NRC establish a suitable process for utilizing expert judgment which is disciplined and carefully structured and allows for interactions between experts.

Another conclusion of NUREG-1150 is that the plants studied meet the safety goals dealing with the risk of prompt and latent cancer fatalities established in a NRC policy statement published in 1986. While this in an important conclusion, it was also found that not all the plants analyzed meet the tentative performance guideline on the overall mean frequency of a large release of radioactive materials to the environment. Surry and Zion overlap to some degree or are below the tentative criterion. Sequoyah, on the other hand, overlaps the tentative criterion but with a substantial portion residing above this criterion.

Careful detail and consideration should be given by the NRC to the definition of a "large release" and the use of this criterion. While all the plants meet the safety goals, they do not meet this tentative criterion. The basis for this tentative criterion as currently defined is too subjective to be of use in the decision making process and it is redundant to the safety goals. A number of factors are important in the measure of risk - including siting, population, etc. Also many modeling factors in PRA analyses can affect how a plant meets the criterion. We suggest that the NRC review the basis for the criterion and whether it is to be applied in the future.

A large body of technical data has been developed by the industry and specifically the WOG which must be incorporated into the probabilistic assessment of Westinghouse NSSS's. All three of the PWRs modelled in NUREG-1150 were analyzed in prior PRAs. The NUREG-1150 plant models assume that a large number of these prior models were appropriate. Significant conservatisms have been identified and changes in plant design and operation have occurred since the original models were developed. These areas include vulnerability to seal LOCAs, development of detailed Emergency Response Guidelines, vulnerability to relief and safety valve LOCAs, procedural verification and validation programs, enhancement of post accident monitoring systems, trip reduction programs, improved station blackout phenomena, application of diesel generator improvement programs, use of non-safety grade equipment in accident mitigation, installation of AMSAC and undervoltage trip coil actuation and numerous assessments which minimize vulnerability to common cause failure modes.

In particular, the seal LOCA quantification model reported in NUREG-1150 appears to be conservative based upon the latest RCP seal PRA analysis, and test results performed for the Westinghouse Owner's Group. This conservatism lead to a much higher importance of seal cooling sequences. In addition, no credit/sensitivity is provided for use of new seal O-ring materials currently being recommended by the WOG.

Although we agree with the overall conclusion of safety, the use of inappropriate models which do not represent the current as-operated, analyzed and maintained plant leads to unjustified emphasis on currently resolved issues. The Westinghouse Owner's Group is willing to participate in an exchange of information, peer review of whatever vehicle is deemed appropriate to incorporate current technical data which would result in a more accurate portrayal of plant and containment operation.

In summary, we feel that some aspects of NUREG-1150 and its supporting documents need to be thoroughly reviewed and revised as necessary. More detail should also be provided in some sections of the report. Furthermore, the modeling conservatisms should be reviewed and more best estimate calculations should be utilized. The Westinghouse Owner's Group has gained expertise in some of these areas and is willing to lend support to the NRC and its contractors so that NUREG-1150 will adequately reflect the present state of the nuclear power industry with respect to severe accidents.

Based on the technical nature of our comments, we recommend that resolution of these comments as well as other comments received from the nuclear industry be adequately addressed prior to final publication of NUREG-1150 and its supporting documents. Additionally, we recommend that the NRC and all nuclear industry representatives work together through some form of open technical exchange meetings to resolve all the concerns that now surround NUREG-1150. These meetings could bring together technical experts to focus their expertise on developing a mutually agreeable solution. We recommend that a united position to the resolution of NUREG-1150 concerns would better serve the goals of the nuclear industry.

WESTINGHOUSE OWNER'S GROUP

COMMENTS ON NUREG-1150

REACTOR RISK REFERENCE DOCUMENT

SEPTEMBER 1987

COMMENTS REPORT

This document details comments regarding the overall methods and the analysis presented in NUREG-1150 for Westinghouse NSSS PWRs. The review is broken down into the plant analysis and the containment and source term analysis comments. Due to the extensive size of the documentation and relatively late availability of supporting documentation during the comment period, the WOG has not identified all issues and comments which would be typical in a peer review process. We have focused on issues for which the WOG could provide direct support based on its ongoing programs and in areas which could significantly enhance the accuracy and validity of the program.

1. CORE DAMAGE FREQUENCY COMMENTS

This section contains comments on the core damage frequency analysis performed in NUREG-1150.

A. "SMART" APPROACH

A "smart" approach, i.e., using detailed fault tree, simplified fault trees, black box models, and/or Boolean expressions as decided upon by the analyst conducting an analysis, was used in the study to evaluate accident sequences leading to core damage. NUREG-1150 reports the "smart" approach as being necessary due to costs and schedule constraints and is justified based upon using highly experienced PRA analysts to conduct the study. A further justification is inferred in the study in that sensitivity studies were conducted on important parameters/issues that are associated with the core damage estimation.

The use of small event/large fault tree methodology to quantify core damage frequency is well recognized in PRA, however using a "smart" approach in the quantification process may yield results that are either pessimistic or optimistic dependent upon how one applies such an approach (i.e., for example, if a conservative approach has been used in a previous analysis and such analysis serves as an input to a "smart" approach application, the end results will repeat being conservative). A very detailed review of NUREG-1150 and supplement reports, paying particular attention to all details of how the "smart" approach was applied in the quantification of the core damage frequency, should be performed to determine if CDF results are pessimistic or optimistic.

If the NUREG-1150 is to provide a basis for future regulatory use and to provide a benchmark for future work, the SMART approach could lead to erroneous prioritizations. Thus, we suggest that the "SMART" approach not be utilized in a document of this magnitude.

B. SEP COMPUTER CODE

Best estimate calculation and sensitivity studies with regards to prediction of the core damage frequency (CDF) requires that major attention be placed upon the verification and validation of both the modeling and computer codes used to determine the frequency. Both the SETS and SEP computer codes have been verified through usage in past PRA studies, however, it is to be noted that the SEP code has one drawback in that only the lognormal distribution is considered for simulation of all variables.

The use of Monte Carlo simulation to solve an expression for an outcome that is dependent on several input variables is well documented both within and outside the nuclear industry. By sampling repeatedly from the assumed joint probability density function of the "X's" and evaluating "Y" for each sample, the distribution of "Y", its mean, percentiles, etc., can be estimated. In such a process, the assumed density function of the "X's" can yield results that are optimistic or pessimistic dependent on the match of the assumed density functions to actual historical data density functions. The core damage frequencies of plants studied and reported in NUREG-1150 hold true only for the case whereby the density functions of the "X's" (i.e., initiating events, hardware failures, common cause failure, operator recovery actions, etc.) are lognormally distributed. A match of historical data density functions to assumed lognormal density functions of the "X's" should be made prior to drawing conclusions about the core damage frequency distributions estimated by use of the SEP computer code as reported in NUREG-1150.

C. INITIATING EVENT (IE) DATA BASE

The accident initiating event data bases used in NUREG-1150 are stated to be plant specific through the use of plant logs, LER's, EPRI reports and EG&G reports. These statements are highly misleading. Although plant specific data is used, virtually all of the "dominant' sequence initiators use generic data, derived frequencies or involve underlying interpretations of initiator definition which significantly affect system success criteria and event progression. In addition, the WOG and several other Industry groups have trip reduction programs which have already demonstrated significant improvements in plant operations while NUREG 1150 was being developed. This should have been factored into the data trending. These are discussed below.

The LOCA frequencies for all three PWRs is generic. Unfortunately the frequencies and definitions of break sizes are inconsistent. Zion's large break frequency is a factor of two (2) higher than the Surry or Sequoyah value. Surry has a very small LOCA (S3) category for seal LOCAs but the Sequoyah and Zion models group seal LOCAs as small LOCAs (S2). The result is that Sequoyah and Zion are a factor of twenty (20) higher in initiating event frequency for S2s. In addition, it is correctly assumed for Surry that seal LOCAs have very long times before recirculation is required, whereas small LOCAs have much shorter action times. Thus Zion and Sequoyah are also penalized by requiring sooner recirculation switchovers. Another factor of concern is the use of 2E-02 per year for the frequency of small LOCAs. The current operating experience base for PWRs clearly shows that LOCAs/Leaks requiring recirculation for mitigation are much more infrequent. A conservative value of 7E-03 per year is typically used.

The dominant sequence for Zion and Sequoyah in NUREG 1150 are initiated by total losses of component cooling water. Again the frequency is stated to be based on the plant specific design, however, a frequency of failure must be derived using the analyst's imposed success criteria and prediction model. The modeling technique, generic piping failure data and success criteria used in the quantification introduce very large conservatism into the results. Time dependent modeling techniques are available for this type of application as well as plant specific assessments of minimum cooling requirements, configurations and recovery strategies for each of the three PWRs. These comments also apply to the modeling of other support functions such as Vital AC and DC, Service Water, Instrument Air and HVAC.

Another area of concern is transient frequencies. All WOG utilities are part of extensive trip reduction programs. In particular, the WOG has developed several major programs. One of these programs developed an assessment of the frequency and causes of transients where feedwater was lost at the initiation or immediately after a trip. This was based on plant surveys. The results differ from the NUREG 1150 values by almost a factor of ten (10). This would drastically reduce auxiliary feedwater and Bleed/Feed challenges.

The above factors at a minimum should be reviewed and incorporated into the baseline calculations in NUREG 1150.

D. SEAL LOCA MODEL

The seal LOCA quantification model reported in NUREG-1150 appears to be conservative based upon the latest RCP seal PRA analysis, and test results performed by Westinghouse for its Owner's Group. The model for the probability of a seal LOCA used in NUREG-1150 is based upon a seal failure having a Weibull probability density function with an increasing hazard rate. In the modeling both worst case and best case probability distributions are defined. The fitting parameters for the Weibull distributions are presented in the appendix of Surry and Sequoyah supplement reports to NUREG-1150. Appendix A of NUREG/CR-4550, Volume 3, Surry, states:

> "Fitting constants for the best and worst case Weibull distributions were based on expert opinion of seal LOCA performance. The question was asked, 'Under the worst (best) conditions of seal performance, at what time do you feel 95% certain that a seal LOCA will (will not) occur?'"

No indication is given by the material presented of just how these parameters were determined. Justification for Weibull parameter selection as input to seal failure flow rates should be documented in the report.

Furthermore, some of the over conservatisms in the modeling should be reduced. The probability of seal LOCA was modelled with a Weibull distribution with 5 percent and 95 percent probabilities corresponding to about 1 hour and 10 hours. Seal failure was defined as complete failure of all three stages. The leak rate was calculated to be 450 gpm.

The Westinghouse Owner's Group concluded in the report "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power" :

> "Assuming the integrity of the secondary sealing elastomers, the results of detailed thermal hydraulic two-phase flow analyses indicated that the leakage flow rate through the RCP seals and support systems would be limited to 21.1 gpm per pump or less. This calculated leakage will not lead to core uncovery for more than 8 hours following a station blackout, therefore providing adequate time for the recovery of seal cooling and makeup capability prior to the beginning of core uncovery."

Furthermore, the report states:

> "The existing seal system components have been shown to have considerable capacity to survive the low probability loss of all seal cooling event, which is beyond the design basis."

The RCP seal LOCA model used in NUREG-1150 should be revised to include this type of information.

E. HARDWARE FAILURE DATA BASE

The hardware data banks used in the NUREG-1150 study are plant specific. Both generic and plant specific hardware failure data were used in sequence cut set quantification. An ASEP generic data base was used in cases where no plant specific data was found. The ASEP data base was generated from a compilation of recent PRA and related studies (ie., EPRI Studies, NUREG-1032, Blackout Studies, Zion, Indian Point, etc.). It is noted as representing a larger experience base for the determination of failure rates and may not apply to any one particular plant. The expressions given in the IREP Procedures Guide were used in the study to estimate component failure probabilities. Error factors are assigned in the study to failure mode failure rates (generic and plant specific). In the Surry supplement report (NUREG/CR-4550, Vol. #3), the data base is shown with error factors at a super-component level while the Sequoyah supplement report (NUREG/CR-4550, Vol. #5) shows a data base with error factors at the component level. The Surry report gives no clear indication of how error factors associated with the component failure probabilities were propagated to the super-component level.

It should also be noted that considerable plant specific data was used in the quantification of CDF for Surry while very little plant specific data was used for Sequoyah CDF quantification. A brief check of both the Surry and Sequoyah component failure rates and associated error factors indicates difference in rates and error factors from those presented in the IREP Procedure Guide and in other PRA studies. The mean values presented are somewhat higher than previously reported and differences can be noted in assigned error factors. The development of the ASEP data base is not presented in NUREG-1150 but is referenced in the supplement report document NUREG/CR-4550, Vol. #1. At the time of the review, Volume #1 of NUREG/CR-4550 was not available for review. A detailed review of the ASEP data base development should be conducted to determine if data reported is conservative or otherwise.

F. TREATMENT OF COMMON CAUSE/DEPENDENT FAILURES

Both generic and plant specific Beta factors were used in accident sequence quantification. Specific ground rules were followed in the application of the Beta factors. One of the ground rules, failure of the third, fourth, etc. redundant components assumed to have a probability of one after failure of a second redundant component due to common cause, has a significant impact, such as in the Sequoyah analysis for the loss of component cooling water special initiator. In the Sequoyah analysis, three CCW pumps (1A, C-S, and 2A) were assumed to be operating and one pump (1B) was in standby. The Beta factor was applied for failure of the C-S CCW pump (given pump 1A has failed). However, this was also assumed to fail the 2A CCW pump due to common cause. This implies that increasing redundancy has no effect on the system unavailability and ultimately, on the core damage frequency. Some justification based on actual operating experience should be provided to support this ground rule, or other methods should be utilized to determine common cause.

The generic component Beta factor data base used in NUREG-1150 was derived by assuming the Beta factor values reported in EPRI document NP 3967 as 95% upper bounds of a lognormal distribution with an error factor of three. The study reports that using the mean values given by EPRI NP 3967 directly, without screening the events used to evaluate them, was felt to be conservative. The significance of the assumption concerning derived Beta factor was evaluated in the study by performing a sensitivity study using the Beta factor values quoted in EPRI NP 3967 as the means.

This simplified method to quantify common cause failures is more conservative relative to other more complicated methods (i.e., Multiple Greek Letter Model, Binomial Failure Rate Model, etc.). Justification for its use is based upon sensitivity studies to determine impact of the generic Beta factors and comparison being made by the analysts with other studies and treatment of common cause failures. As the treatment of common cause failures has a direct impact in the quantification of CDF, a more exact method of treatment is required to obtain a better estimate of CDF's and their uncertainties than reported in the study.

G. TREATMENT OF HUMAN ERROR PROBABILITY (HEP)

Human error probability values used for sequence cut set quantification were derived using a simplified human reliability analysis (HRA) developed for ASEP. Both the pre-accident and post-accident screening procedures were developed with the intent of being deliberately conservative (i.e., pre-accident HEP set at .03 with little credit for recovery and post-accident HEP set at 1.0 for all actions outside the control room and .05 for critical post diagnosis task in the control room).

A review of both supplement reports (Vol. #3 and Vol. 5, NUREG/CR-4550) for Surry and Sequoyah did not reveal if the screening procedures for HRA developed for ASEP was employed. It appears, based upon a brief review of these reports, that only nominal HRA procedures were employed. The screening procedures should also include utilizing best estimate probabilities and response times.

A brief review of the treatment of HEP in NUREG-1150 revealed that the results are conservative in several areas such as: (1) no credit for operator actions not explicitly stated in plant procedures, (2) recovery actions constrained by the established HRA ground-rules, and (3) HEP screening values included in HRA ground-rules. This is particularly evident given the numerous procedures and alternate equipment available for alignment to minimize the effects of the accident. For example, credit was not taken for crossties in the Sequoyah analyses of the component cooling water system.

In addition, an error factor of 10 was assigned to assumed HEP valves used in quantification. This is conservative based on the HEP uncertainty given in the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (NUREG/CR-1278). The error factors in Table 20-20 of NUREG/CR-1278 range from 3 to 5. As the analysis of HRA directly impacts CDF and its uncertainty it should be closely scrutinized in all details prior to drawing conclusions about obtained CDF's.

2.0 CONTAINMENT AND SOURCE TERM COMMENTS

This section contains comments on the containment and source term analysis performed in NUREG-1150.

A. USE OF EXPERT JUDGMENT

Expert judgements as opposed to mechanistic calculations were used to quantify the impact of many important phenomena. In general, these judgements govern the environmental releases used in the study and therefore the conclusions. In most cases it is stated that this was done since models for these phenomena do not exist in the Source Term Code Package (STCP). Some of the more important cases are:

a. Revaporization of fission products after vessel failure.
b. In-vessel natural circulation and induced LOCAs in the steam generators and hot leg.
c. Natural circulation ex-vessel.
d. Direct containment heating.
e. Chemical form of fission products, especially iodine.
f. Steam explosions sufficient to fail containment.

In some other cases, expert judgements were used even though STCP models exist. For example, judgement was used to assign pressure increments for hydrogen burns. In all of these cases, there appears to be no supporting calculations to justify the experts' opinions, nor is there even documentation to describe the reasoning by which the probabilities were estimated. In general, the fact that results on the extremes of the distribution are often assigned high probability tends to raise questions about the methodology. The experts themselves described many of the judgements as "guesses." This situation is particularly distressing since these unquantified, undocumented opinions dictate the source term characterization. Furthermore, in many areas sufficient information exists to make such judgements unnecessary. For example in the first three items, there exists detailed models, experimental data, benchmarks of the models, and even simple hand-calculational methods in some cases. These do not even appear to be considered. In addition, at the time of the study detailed models were not yet incorporated in the integrated severe accident codes for the last three issues, but a wealth of data generally exists. In light of this data, the probability ranges assigned to direct heating and stem explosion-induced containment failure in particular appear too high.

In summary, because of the importance of the source term characterization we recommend that the use of expert judgements be confined strictly to areas where models and benchmarks do not exist. For those few areas where this is true, it is essential to

a. Choose "experts" with directly applicable experience. This is essential if the range of uncertainty is to be assessed reasonably.
b. Choose experts from industry and academia as well as the national laboratories and NRC.
c. Ensure that the stated positions of the experts are supported by both analyses and comparison to data.

This is crucial if the study is to be a state-of-the-art, scrutable representation of the risk related to nuclear power plants. It is our technical opinion that the use of judgment in the current document grossly overstates the environmental releases in the case of severe accidents at a U.S. nuclear plant and therefore destroys the usefulness of the document in decision making.

B. USE OF PARAMETRIC SOURCE TERM MODEL

In order to supplement and extrapolate the STCP calculations, non-mechanistic parametric source term models were developed and exercised. It is not at all clear from the publicized documents that these calculations can be supported. At the very least, detailed descriptions of these models, how the input data were obtained, and comparisons of their predictions to STCP calculations should be provided.

C. NEGLECT OF IMPORTANT OPERATOR ACTIONS

A review of the containment response and source term assessments revealed what appears to be a significant omission, namely, the neglect of many human actions in the NUREG-1150 quantifications. Key human actions have been shown to have major and beneficial impact on reducing the risk of operating nuclear reactors. Some of the more important in-place procedures (FRGs) which have been shown to greatly mitigate severe accidents and which are neglected in NUREG-1150 include:

a. Manual operation of the AFW system in blackout events. In the case of the NUREG-1150 Zion study, sequence 1 assumes core melt will occur one half hour after seal LOCA. However, if turbine-driven AFW is available, a steam generator reflux cooling mode could be established which would enable use of the accumulators and delay core melting for 10 hours or more. This is a significant interval of time which could allow for recovery and accident termination or depressurization of the primary system before core melt and hence the elimination of postulated direct containment heating effects.

b. Refilling the reactor water storage tank (RWST) which would provide for prolonged make-up to the primary system if recirculation was not functional.

c. Opening the pressurizer PORVs when core outlet temperatures exceed 1200°F would remove all concerns for steam generator tube integrity and lessen residual concerns for direct containment heating.

Neglect of such key operator actions distorts the evaluation and undermines the usefulness of the document.

D. LARGE DRY CONTAINMENTS

In the Zion study the failure of CCWS piping is the major contributor to sequence 1 which represents the dominant core melt sequence. However, this result is based on the unsupported and grossly conservative assumption that the piping catastrophic failure rate for high pressure systems as developed in WASH-1400 also applies to the low pressure piping of the CCWS. This assumption is tantamount to assuming that this is a new dominant sequence for Zion. It is not enough to simply treat this failure rate in the uncertainty assessment as it can skew the results and mask other findings. Since this effort is meant to rebaseline risk assessment and update the previous work, effort should be made to improve the subject failure rate data before employing it in NUREG-1150.

E. ICE CONDENSER CONTAINMENT LOADS

The Sequoyah containment performance analysis carried out in NUREG-1150 considered ten different containment loads [1]:

1. hydrogen burn prior to vessel failure,
2. hydrogen burn at vessel failure,
3. mode of reactor vessel breach,
4. direct containment heating,
5. induced failure of the reactor coolant system as a result of structure overheating,
6. formation of a coolable debris bed,
7. impairment of the ice condenser function due to ice depletion or a detonation,
8. direct contact of debris with the containment shell,
9. containment failure by overpressure (60, 75 and 90 psia), and
10. failure of the containment spray due to a major containment failure.

Discussion of some of these key loads are discussed below.

E.1 Hydrogen Burns

In NUREG 1150, hydrogen burns and detonations were stated to place a severe load on the containment that could lead to its failure. This load depends on the availability of ice, igniters and fans during the accident. At vessel failure, hydrogen burns generating peak pressures of 100 psia were considered.

The IDCOR approach to in-vessel hydrogen generation and hydrogen burn in the containment was presented to the NRC as part of the Technical Support for Issue Resolution (Reference [8]), Issues 6 and 16. In essence the IDCOR models predict smaller amounts of hydrogen generation during in-core oxidation and represents the performance of the igniter system (in terms of number and location of igniters) to allow for local burning of the hydrogen. In the case of a station blackout, where the igniter systems are not available, the hot debris in the cavity and natural circulation in the containments will result in recombination of the combustible gases produced in the cavity without a severe load on the containment shell. NUREG 1150 results do not appear to adequately credit the effects of the igniters, ignore recombination, and have very large hydrogen source terms.

E.2 Reactor Vessel Breach

Steam explosions in the lower plenum are considered in NUREG-1150 as a possible mechanism for failure of the reactor vessel and for the , failure mode of the containment. This is in contrast to the conclusion of the NRC review group on steam explosions. As stated in the resolution of Issue 7 (Reference [8]), IDCOR agreed with this conclusion.

Further, MAAP analyses have shown that temperatures in the hot leg will greatly exceed steam generator tube temperatures. Since failure temperature of the hot leg is similar, even when uncertainties are considered other parts of the primary system would be expected to fail long before the tubes. This would depressurize the primary system and remove all concerns for tube integrity.

E.5 Formation of Coolable Debris

The NUREG-1150 assigns a 50% probability for the corium to be quenched by water, and considers the possibility that the water would not scrub the fission products that are released from the debris. Debris coolability was the subject of IDCOR report 15.28 [11] which concluded that debris will be cooled if covered by water at a rate similar to that given by the pool boiling critical heat flux. Also, if the overlying water pool is sufficiently deep, high decontamination factors will be achieved (Reference [12]).

E.6 Containment Failure Model

A major failure of the containment which leads to a large failure area is considered in NUREG-1150. IDCOR position on this issue is given in Reference [8], which states that leak-before-break is the likely failure mode of the containment as it is overpressurized by internal processes.

E.7 Containment Event Tree

As a result of the differences in containment loads discussed above, the containment event tree analyzed in the NUREG-1150 contains many more branches that the IDCOR event tree for Sequoyah. The former (NUREG-1150) contains 25 bins that characterize the containment performance during severe accidents. Sixteen bins (out of 25) are due to early containment failure caused by direct containment heating and hydrogen burns and detonations. Three other bins are associated with induced steam generator tube rupture leading to containment bypass. Only the remaining six bins are due to internal loads that are consistent with the IDCOR position and the results of the issue resolution effort. This illustrates the importance in NUREG-1150 of failure modes considered highly unlikely by IDCOR.

F. SOURCE TERM RESULTS

Even under late containment failure conditions, many energetic events are considered in the uncertainty of source term releases in NUREG-1150. This uncertainty lead to a high central value of the source term releases. Comparisons of the source terms from similar end states are shown in Table 1. Although differences in relative terms are noted, especially with respect to Te, Sr and Ru (Mo), in absolute terms the releases in all the end states with a functional containment are rather small and mostly dominated by noble gases.

The largest differences in the source term are in the case of late dry releases (state IIa and bin 19). This is believed to be partially due to the timing of containment failure (30 hours in the IDCOR analysis while for NUREG-1150 the time is just a few hours after vessel failure). Also, as mentioned in Section E.1, differences in modeling hydrogen burns is the reason for the larger source term in bin 19.

This comparison illustrates that when similar sets of phenomena are considered, estimates of source terms calculated by the NRC and IDCOR are usually consistent.

Table 1

COMPARISON BETWEEN CENTRAL ESTIMATES OF RELEASE FRACTION FROM REFERENCE [2] (BASIS OF NUREG-1150) AND IDCOR IPE [7]

Fission Product Group	Containment Not Failed		Late Containment Failure, Debris Scrubbed (wet)		Late Containment Failure, Debris Not Scrubbed (dry)		Large Isolation Failure	
	IDCOR I	NUREG Bin 23 (19)	IDCOR IIa	NUREG Bin 21 (17)	IDCOR IIb	NUREG Bin 19 (15)	IDCOR IV	NUREG Bin 15 (11)
Noble Gases	8E-4	5E-3	0.91	1.0	0.5	3E-2*	0.98	1.0
CsI	< E-5	3.4E-5	$2.3 \cdot 10^{-4}$	5.0E-3	1.2E-4	1.7E-2	3.3E-4	9E-3
CsOH	< E-5	< E-5	$6.8 \cdot 10^{-4}$	5.0E-3	2.5E-4	1.7E-2	3.4E-4	4.3E-3
Te	< E-5	< E-5	< E-5	5.0E-4	1.1E-4	4.1E-2	< E-5	1.7E-2
Sr	< E-5	2E-5	< E-5	5.0E-4	3E-5	9.4E-2	< E-5	1.3E-2
Ru (Mo)	< E-5	1.0E-5	< E-5	1.1E-4	7E-5	5.7E-2	4.9E-5	1.3E-2

*This is probably due to a typing error in Table 5-6, Reference [2]. The correct value should be ~ 0.97.

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