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SUBJECT: Comment on NUREG 1150 re reactor risk ref document.
Recommends NRC utilize listed info to correct technical
deficiencies in current draft of NUREG-1150.

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U.S. Nuclear Regulatory Commission
Office of Administration
Division of Rules and Records
Washington, D.C. 20555

Attention: Chief, Rules and Procedures Branch

COMMENTS ON DRAFT NUREG-1150 (REACTOR RISK REFERENCE DOCUMENT)

Enclosed are TVA's comments to draft NUREG-1150 in accordance with the Federal Register notice volume 52, No. 49.

The comments are divided into three sections.

Enclosure 1 is a presentation type overview; enclosure 2 is an executive summary; and enclsoure 3 contains detailed comments.

TVA does not plan to provide further comments to NUREG-1150; however, should additional review identify a need for additional input, TVA will provide the information in time to support the overall October 1, 1987 schedule for industry comments on NUREG-1150.

No commitments are contained in this correspondence.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R Gridley

R. L. Gridley, Director
Nuclear Safety and Licensing

Enclosures
cc: See page 2

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U.S. Nuclear Regulatory Commission

SEP 28 1987

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SUMMARY OF
COMMENTS ON NUREG-1150 REACTOR RISK REFERENCE DOCUMENT

- NUREG-1150 fails to consider significant industry sponsored research and analysis.
- The lower bound of NUREG-1150 probability estimates exceeds comparable Industry Degraded Core Rulemaking (IDCOR) values.
- NUREG-1150 overstates the risk because of early containment failure.
- NUREG-1150 overstates the magnitude of fission product releases.
- NUREG-1150 does not adequately represent the ability of operators to terminate or mitigate accidents.
- NUREG-1150 relies heavily on expert opinions of a limited and select group of individuals which included no utility or vendor participation and which is not adequately documented.
- NUREG-1150 uses inadequate Sequoyah Nuclear Plant (SQN) systems models principally because of inappropriate analysis assumptions.
- SQN Individual Plant Evaluation (IPE) and IDCOR studies demonstrate significant over-conservatisms in NUREG-1150 methodology.

ENCLOSURE 2

OVERVIEW OF NUREG-1150

1.0 Impact of NUREG-1150 on SQN

1.1 Current NRC Safety Goals

SQN meets the current NRC safety goals.

1.2 Proposed NRC Safety Goals

SQN may not meet the new proposed safety goal secondary guideline of a large release frequency of less than 1×10^{-6} per year based on the results of NUREG-1150.

2.0 Major Issues

2.1 Early Containment Failure

2.1.1 Probability of Early Containment Failure

NUREG-1150 predicts that the probability for early containment failure is 0.02 to 0.50 given a core melt sequence.

2.1.2 IDCOR/TVA Position on Early Containment Failure

IDCOR predicts no early containment failures except for the V-sequence (interfacing LOCA) and containment isolation failure which have a low probability of occurrence.

2.2 Early Containment Failure Mechanisms

2.2.1 Hydrogen Generation, Transport, and Combustion

Hydrogen generation, transport, and combustion as modeled in NUREG-1150 is highly conservative resulting in exaggerated early containment failure frequencies particularly for ice condenser containments.

2.2.2 Direct Containment Heating

Direct heating of the containment atmosphere, although considered viable in NUREG-1150, was considered by the IDCOR studies but was found not to be a contributor to containment failure particularly for the SQN cavity design.

2.2.3 Steam Generator Tube Failure

Induced failure of the steam generator tubes because of high temperatures, included in NUREG-1150 as a contributor to early containment failure, was not considered credible in the IDCOR studies.

2.2.4 Debris Bed Cooling

Debris bed cooling by overlying water pools and the decontamination factors associated with fission product release from the debris is modeled very conservatively in NUREG-1150.

2.3 Source Term Calculations

The NUREG-1150 source terms are highly dependent on the timing of the predicted containment failures. For those sequences that are considered classically risk dominant, the NUREG-1150 analyses predicted much higher source terms because of early containment failure than did the IDCOR analysis. This great discrepancy in containment failure times is directly attributable to those assumptions in the NUREG-1150 analyses that lead to early containment failure discussed previously in section 2.2.

2.4 Offsite Consequences

The NUREG-1150 analyses used the NRC's computer code MELCOR consequence module MACCS and CRAC2. The IDCOR analyses (References 3 and 7), which used CRAC2, predicted significantly lower offsite consequences than did the NUREG-1150 analyses.

2.5 Systems Modeling

Major discrepancies have been identified in the systems assumptions which impact the core damage frequency as reported in NUREG-1150. Some of these are also confirmed in the SQN Individual Plant Evaluation (IPE) (Reference 8). The following major assumption differences include:

2.5.1 Centrifugal Charging Pump Dependence on Component Cooling Water

NUREG-1150 assumes that the total loss of Component Cooling Water (CCS) leads directly to a loss of centrifugal charging pumps which is in direct disagreement with recent tests conducted by Westinghouse.

2.5.2 Reactor Coolant Pump Seal Loss of Coolant Accident (LOCA) Model

The assumed flowrate, given a Reactor Coolant Pump (RCP) seal LOCA, is very conservative from a flowrate standpoint and does not reflect the latest seal leakage test results conducted by the Westinghouse Owners Group (WOG).

2.5.3 Initiating Event Frequency

The initiating frequencies for frequent and infrequent events are conservatively high based on generic rather than plant specific and industry experience used by the SQN IPE.

2.5.4 Common Cause Analysis (Beta Factors)

The core damage frequency analysis used in NUREG-1150 incorrectly applied the beta factors presented in EPRI NP-3967 (Reference 10). The beta factors were applied to systems at SQN in such a way that negated much of the redundancy and flexibility designed into these systems.

2.5.5 Pessimistic Success Criteria

The success criteria assumed in the NUREG-1150 analysis is extremely conservative. In numerous instances, this conservative success criteria resulted in core damage sequences which are not predicted using the most recent assumptions on success criteria.

2.5.6 Containment/Consequence Analysis

NUREG-1150 utilized limited plant specific containment analyses resulting in an unrealistic portrayal of SQN leading to conservative assumptions in the containment consequence analysis.

2.5.7 Adequacy of Event/Systems Analysis

The event analysis lacked a detailed assessment of the realistic plant response to the most important sequences thus invalidating the risk reduction analysis.

2.6 Uncertainty Analysis

2.6.1 Risk Quantification

IDCOR risks are below the lower bounds of the NUREG-1150 risk range. This indicates the uncertainty analysis may be biased.

2.6.2 Basis for Uncertainties

The uses of expert opinion are not carefully and completely documented in NUREG-1150. This needs to be done for a full assessment of the technical validity of the opinions and for the benefit of experts who may have differing opinions.

2.6.3 Calculation Method

The statistical methods used in the uncertainty analysis resulted in a significant upward shift of the mean value of the plant damage states for the dominant sequences.

2.7 Operator Actions

The modeling of operator actions in the NUREG-1150 analyses was overly conservative. Many operator recovery functions were not modeled.

2.8 Risk Estimation

Comparing the central point risk estimates between NUREG-1150 and IDCOR, the NUREG-1150 analyses indicate significantly higher risks in comparison to the IDCOR analyses.

2.9 Calculation of Risk Reduction

The calculation of the effects of various preventative and mitigative options has on risk reduction is at best misleading. The risk reduction analysis assumes that: (a) "each option functions properly as designed," (b) "the option under consideration was designed, installed, and operated properly," and (c) "did not interfere adversely with other plant features in a subtle and unknown way." The combination of these assumptions and limitations of the NUREG-1150 analysis resulted in an analysis which for several options has apparently calculated the effect of changing the NUREG-1150 analysis assumptions rather than the effect of plant modifications.

ENCLOSURE 3

DETAILED NUREG-1150 COMMENTS

1.0 Impact of NUREG-1150 on SQN

1.1 Current NRC Safety Goals

On page ES-10 of Reference 1, Volume 1, it is stated that all the reference plants, including SQN, meet the explicit risk criteria stated in the NRC's safety goal policy.

1.2 Proposed NRC Safety Goals

A revised criterion is under consideration by the NRC that would require the "overall mean frequency of a large release from a reactor accident should be less than one in a million per year." A large release has been defined as one that is sufficient to cause one or more early fatalities. SQN may not meet this new criterion since a substantial portion of SQN's uncertainty band lies above the proposed goal (see Figure ES.7 of Reference 1, Volume 1). In fact, SQN shows the worst results for this criterion of those reference plants analyzed according to NUREG-1150.

Using the draft NUREG-1150 results, the current NRC safety goals are satisfied by SQN. However, future goals may not be met. SQN may not meet the new proposed safety goal of a large release (resulting in one or more early fatalities) frequency of less than 1×10^{-6} per year if NUREG-1150 as it stands today is used as the decisionmaking tool.

2.0 Major Issues

2.1 Early Containment Failure

2.1.1 Probability of Early Containment Failure

NUREG-1150 reports the probability of early containment failure, given a core melt has occurred, ranges from 0.02 to 0.50 (see Figure ES.11 of Reference 1, Volume 1). Major disagreements still exist between the containment response controlling phenomena as modeled in the IDCOR program and those analyses performed in support of NUREG-1150. These disagreements have been identified and well documented (see Reference 2) during the past several years as part of the closure activities between IDCOR and the NRC reference plant analyses. Therefore, rather than reiterate all of these issues, specific problem areas will be enumerated below.

2.1.2 IDCOR Position on Early Containment Failure

The IDCOR reference plant analysis, as reported in Reference 3, predicts no prompt containment failures except for the V sequence and containment isolation failure which have a very low probability of occurring. In contrast, NUREG-1150 allows for the possibility of a variety of mechanisms leading to early containment failure including hydrogen combustion, direct heating, and induced failure of steam generator tubes.

Late containment failure would result in less fatalities than early containment failure because of:

1. increased warning time resulting in greater time being available for evacuation of the surrounding populace, and

2. increased effectiveness of fission product deposition processes along with any decay which may occur during this period.

It should also be noted that the NUREG-1150 reviewers depended on past studies or simplistic calculations for determining the expected containment response instead of depending on SQN specific calculations. This approach may be valid for evaluating a large dry containment. However, because of the complex design of the ice condenser including compartmentalization and dynamic response, such approximations are virtually meaningless and provide misleading results for SQN.

2.2 Early Containment Failure Mechanisms

2.2.1 Hydrogen Generation, Transport, and Combustion

Both the IDCOR and NRC reference plant analyses recognize the fact that hydrogen generation, transport, and combustion within the ice condenser containment is a major factor in predicting the containment response to degraded core accident sequences. However, basic fundamental differences in the phenomenology still exist between the industry and the NRC. Whereas, the industry predicts the controlled combustion of hydrogen within the ice condenser containment for cases with hydrogen igniters available, the NRC still predicts hydrogen generation and combustion rates high enough to challenge containment integrity. This major difference in containment response can be traced to those unresolved IDCOR/NRC issues reported in Reference 2.

The hydrogen combustion issue is a key issue for early containment failure. As mentioned above, Reference 2 (issues 6 and 16) and presentation materials provided by IDCOR and the NRC during the issue resolution closure meetings document the major differences between the phenomena modeled in the analyses. Some of the major differences noted are the rates of hydrogen generation both in-vessel and ex-vessel, consideration of steam inerting, modeling of incomplete combustion, effects of hydrogen recombination, etc.

The IDCOR analyses predict lower release rates in-vessel than does NUREG-1150 because of different assumptions concerning the core geometry, material displacement, and debris fragmentation during the meltdown progression. The issues surrounding hydrogen ignition and combustion are very complex because of the various dependencies on the concentrations of hydrogen, oxygen, and steam; ignition limits and sources; availability of ice; and spatial relationships between the various compartments and ignition sources. The IDCOR analyses predict that natural circulation in the containment coupled with the hot debris in the cavity will result in recombination of the combustible gases produced in the reactor cavity, thus reducing the total amount of hydrogen available for combustion even in cases without igniters available. Therefore, stagnant regions rich in hydrogen were minimized in the IDCOR analyses unlike the NUREG-1150 analyses which resulted in the formulation of detonatable hydrogen concentrations.

It should also be noted that the issues, such as ignitor design, surrounding hydrogen ignition were extensively studied for the SQN containment by both TVA and NRC as part of the initial licensing of the plant. Analysis and research have shown that the igniters are far more effective than judgement would have predicted.

2.2.2 Direct Containment Heating (DCH)

The containment load associated with DCH is postulated in NUREG-1150 to be a major cause of early containment failure. The DCH issue was also addressed in the IDCOR reference plant analyses. IDCOR determined that the plant-specific geometry representing the SQN cavity design does not allow for a debris dispersive event leading to DCH for the case of a high pressure melt-through of the reactor vessel. This finding is summarized in Reference 2, Issue 8, and in the IDCOR Technical Report 15.2B (Reference 4).

Experiments carried out at Fauske and Associates, Inc. (FAI) on a small scale mock-up of the SQN cavity with simulant materials, have shown that more than 90 percent of the injected melt remains inside the cavity. In addition, hand calculations in Reference 5 indicate that the seal table would not melt because of corium impingement. Instead, the corium would fall to the bottom of the cavity before any steel melting would take place. Therefore, containment failure by this mechanism is not considered credible by IDCOR. This also rules out the possibility of containment failure by direct contact of the shell by the debris. Again, the issues surrounding the DCH have been well documented in the work performed in the IDCOR program.

2.2.3 Steam Generator Tube Failure

The loss of containment integrity as a result of steam generator tube thermal attack and failure from high temperature in the primary system is predicted by NUREG-1150. The NRC report predicts temperatures great enough to result in primary system breach prior to reactor vessel failure. One of the areas of failure considered in NUREG-1150 is steam generator tube failure. This is important because of the possibility of a direct release path which bypasses the containment. However, the IDCOR assessments do not predict failures similar to those in NUREG-1150. The IDCOR results indicate transport and effective deposition of fission products in the steam generators for cases with auxiliary feedwater available. For cases with no feedwater available, the primary path for fission products is through the pressurizer and surge line for high pressure transient sequences. Therefore, the consequential failure of steam generator tubes and bypass of the containment is not considered a credible release path.

2.2.4 Debris Bed Cooling

Debris bed cooling by overlying water pools and the decontamination factors associated with fission product release from the debris is modeled very conservatively in NUREG-1150. In the NUREG-1150 analysis, debris bed coolability had only a 1 in 2 chance of occurring. Additionally, any overlying water above the debris was assumed not to scrub the fission products released from the debris during core-concrete attack. Debris coolability was the subject of IDCOR Technical Report 15.2B (Reference 4) which concluded that debris will be cooled 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 6).

In summary, it should be noted that IDCOR and NRC identified and documented these areas of disagreement that required resolution. Following the identification of these issues, IDCOR addressed these issues and others in the IDCOR Technical Report 85.2 (Reference 2) through additional calculations and experimental support where applicable. None of these issues changed the IDCOR conclusion that early containment failures are very unlikely. A similar report by the NRC that addresses each issue and provides supporting evidence for the NUREG-1150 results would be most helpful and necessary before any of these issues can be credited or discounted. Unfortunately, it appears the NRC would rather assign a very large unrealistic, uncertainty band on the probability of these controlling phenomena than evaluate the issues based on supporting calculations and experimental data.

2.3 Source Term Calculations

The source term provides the comparison measure by which the severity of various accident sequences may be evaluated. The source term refers to the amount and timing of the release of radioactive materials to the environment. Release timing is affected by the time at which the containment fails which in turn is affected by those issues discussed in sections 2.1 and 2.2.

NUREG-1150 compared the predicted release fractions with those used in the IDCOR analysis for the V-sequence and a blackout sequence (see Reference 1, Volume 1, pg 5-18). Examination of this comparison shows the IDCOR release fractions for the V-sequence are within the NUREG-1150 uncertainty bands. However, for the blackout sequence, a significant difference exists because of the IDCOR prediction of a late containment failure (on the order of 27 hours) compared to an early containment failure predicted by NUREG-1150 to occur in a few hours following vessel breach.

Therefore, it appears that the differences between source term predictions for IDCOR and NUREG-1150 are primarily affected by the mechanisms of containment failure. Another difference that may be important is the core-concrete interaction fission product release models used in IDCOR as compared to the NUREG-1150 study. For example, it was pointed out on page 5-18 of Reference 1, Volume 1, that this difference leads to IDCOR predicted strontium releases which fall at the bottom of the NUREG-1150 strontium release range.

2.4 Offsite Consequences

The offsite consequences were predicted using the MACCS and CRAC2 computer codes developed as part of the MELCOR program. The offsite consequences reported in the IDCOR studies utilized the CRAC2 computer code. The MACCS code was developed to incorporate certain improvements in the health effects calculations and the ability to model time dependent releases. These two areas of improvement are needed to provide more realistic modeling of release profiles. However, as with any new computer code, a "shakedown" period of use has revealed two problem areas (see Reference 1, Volume 1, pg 6-3). A comparison of predicted consequences between the IDCOR and NUREG-1150 analyses is presented in Reference 1, Volume 1, pg 6-13. For a spectrum of accident sequences, the NUREG-1150 analyses indicates 0 to 10 early fatalities and 10 to 100 early injuries. In contrast, the IDCOR estimates showed no early fatalities and the long-term consequences were significantly lower than those reported in NUREG-1150. These apparent differences were attributed to the much lower source term as predicted by the IDCOR analyses. It should be noted, however, that the IDCOR results were based on the assumption of 100 percent evacuation of the population. This assumption and more recent IDCOR consequence analyses are discussed in more detail in the following paragraph.

The assumptions concerning the site specific meteorology, population distributions, and evacuation strategies are very important in assessing the offsite consequences. According to Reference 1, Appendix D, page D.8, the meteorology and population distributions used in the analysis are site specific. However, one area of disagreement was discovered in the evacuation assumptions as reported in Reference 1, Appendix L, page L-37. It is stated in this appendix that the IDCOR analysis assumed that all, or 100 percent, of the population was evacuated which leads to the reduced health effects in the IDCOR analysis. This statement needs further clarification. The original IDCOR consequence analyses, as reported in Reference 3, assumed that the total population surrounding the plant site was evacuated. However, following discussions with the NRC during the NRC/IDCOR interaction meetings, IDCOR performed a follow-up consequence analysis (Reference 7) for the reference plants that assumed that 95 percent of the population were evacuated and 5 percent were not evacuated. Therefore, the NUREG-1150 and updated IDCOR results both have the same assumption of 95 percent evacuated and 5 percent not evacuated. However, upon comparison of the revised IDCOR health effects with that of NUREG-1150, the IDCOR predictions of health effects are still significantly lower than those reported in NUREG-1150.

2.5 Systems Modeling

Major differences have been identified in the systems assumptions which impact the core damage frequency as reported in NUREG-1150. Some of these are also confirmed in the SQN IPE (Reference 8). The following conservative assumptions are used in NUREG-1150:

2.5.1 Centrifugal Charging Pump (CCP) Dependence on Component Cooling Water

The reactor coolant pumps (RCPs) are dependent on the CCS to provide cooling to the thermal barrier and the motor lube oil. In addition, the CCPs provide seal injection to cool the RCP seals. NUREG-1150 and some previous PRAs model a total loss of CCS as resulting in an RCP seal LOCA. For plant designs where CCP oil coolers, room coolers, and seals are all cooled by CCS, a total loss of CCS would result in failure of the CCPs. For the SQN design, the oil coolers and room coolers are supplied cooling water by the ERCW. Even though the CCP's seals are cooled by CCS it has been shown by actual pump seal tests (Reference 8, Event Trees Notebook, pg 11-1) that the total loss of seal cooling will result in a maximum seal leakage rate of less than 1.5 gpm. Therefore, the total loss of CCS would not result in a loss of function of the CCPs.

2.5.2 Reactor Coolant Pump Seal LOCA Model

In the unlikely case of total loss of CCS and failure of CCPs (e.g., station blackout), it is postulated that thermal overload of the RCP seals will occur leading to a consequential RCP seal LOCA.

The probability of a RCP seal LOCA is modeled in NUREG/CR-4550 (Reference 11) as a Weibull distribution with 5 percent and 95 percent probabilities corresponding to about 1 hour and 10 hours. Seal LOCA flowrate was assumed to be 1800 gpm total (450 gpm per pump). The seal LOCA model used in NUREG-1150 is not consistent with the latest RCP seal analysis and test results performed for the WOG (Reference 9). The WOG seal leakage analysis indicated an expected leakage of 21 gpm per pump at 2500 psia and 550 degrees Fahrenheit. The test results confirmed this analysis with 20 hours of successful test at a stable leakage rate of 16 gpm at 2,278 psia and 534 degrees Fahrenheit. Leakage rates for loss of AC power and cool down conditions were 13 gpm (1,323 psia, 534 degrees Fahrenheit) and 9 gpm (588 psia, 455 degrees Fahrenheit), respectively.

This leakage rate is much less than the base case assumption of 1,800 gpm and is extremely significant as indicated by sensitivity study 1 presented in Reference 11, NUREG/CR-4550, which reduces the total core damage frequency by 5×10^{-5} (45 percent). TVA believes the RCP seals will last longer under loss of cooling conditions (3-6 hours) and failure of the seals will result in a limited leakage of approximately 20 gpm.

2.5.3 Initiating Event Frequency

The initiating event frequencies used in the NUREG-1150 analysis tend to be higher than those reported in the SQN IPE (Reference 8). This is most likely because of the fact that for frequent initiating events (e.g., transients) the NUREG-1150 analysis relied on generic data. In contrast, the SQN IPE based these frequencies on SQN plant specific operating history which included Potential Reportable Occurrences (PROs), Licensee Evaluation Reports (LERs), WOG Trip Reduction and Assessment Program (TRAP) input data, and NUREG/CR-3862 (Reference 12). Similarly, the initiating event frequencies for those infrequent events (e.g., LOCAs) used in NUREG-1150 also tended to be higher than that used in the SQN IPE.

The following discussion indicates areas where the assumed initiating frequencies are considered too high.

A. Small LOCAs (S_2) - The frequency of small LOCAs used in NUREG-1150 is 2 x 10^{-2} per year. This frequency was not developed independently, but was referenced from a previous probabilistic risk assessment (PRA). Current information such as the SQN IPE, leads to the calculation of a small LOCA frequency in the range of 7 x 10^{-3} per year.

B. Loss of offsite power (T_1) - The NUREG-1150 frequency used for loss of offsite power (7 x 10^{-2} per year) was taken from NUREG-1032 (Reference 13) assuming cluster 7 is applicable to SQN. Using SQN-specific information in the evaluation criteria yields cluster 2 which has an initiating frequency of approximately 2.5 x 10^{-2} per year which is the value in the SQN IPE.

C. Nonrecoverable loss of 125 Vdc vital battery board "X" (T_{DCX}) - The calculation of the NUREG-1150 frequency used for this event (9 x 10^{-4} per year) is not well documented in the analysis. However, TVA has reviewed the failure modes included in the reference document for this initiating event (NUREG-0666, Reference 14). The listed failure modes are recoverable. Therefore, a suitable recovery factor for the battery boards should be included for this initiating event.

D. Loss of power conversion system (T_2) - The NUREG-1150 frequency for transients involving the loss of the feedwater system (3.0 per year) is derived from NUREG/CR-3862 (Reference 12) and includes transients in which the main feed pumps are tripped. Tripping the main feed pumps does not result in nonrecoverable loss of the feedwater system. SQN-specific information has been used to develop a frequency of transients with main feedwater unavailable of approximately 0.4 per year (Reference 8). Although this particular frequency had little effect on the analysis results, an adjustment should be made to account for plant-specific data.

E. Loss of component cooling water (T_{CCW}) - The NUREG-1150 frequency for loss of all component cooling water (2.7 x 10^{-5}) is calculated using a beta factor for the failure of all three operating pumps coupled with a failure of a fourth pump. The fifth pump was assumed to be required for unit 2. The frequency of T_{CCW} is overestimated because of several causes:

   1. Beta factors are developed primarily for standby components (the use of generic beta factors for operating equipment is inappropriate)

   2. A beta factor is only applicable to the second redundant component, the third component should not be assumed failed

   3. No recovery is allowed for any of the pumps, operating or standby, nor is credit taken for realignment of the remaining pump to serve important loads for both units

F. Interfacing LOCA (V) - The NUREG-1150 frequency for interfacing LOCA is overestimated because of the inclusion of a combined valve transfer open and valve rupture failure mode. As stated in NUREG/CR-4550 (Reference 1), Volume 5, the valve failure mode "transfer open" can only occur to a check valve which has very little pressure differential across it. Since the procedure for testing check valves when raising RCS pressure ensures both valves are sealed, this failure mode does note apply to check valves 63-560, -561, -562, and -563 since they experience a high differential pressure. If at some later time leakage occurs which reduces this pressure differential, rupture of the upstream valve would restore the high pressure differential necessary to seal the flow that cannot be accommodated by the residual heat removal (RHR) relief valves. In any case, analyses performed by IDCOR show that catastrophic failure of the RHR piping has a very low probability of occurrence. It is more likely that the RHR pump seals would fail resulting in a leakage of primary coolant into the RHR pump rooms.

2.5.4 Common Cause Analysis (Beta Factors)

The core damage frequency analysis used in NUREG-1150 incorrectly applied the beta factors presented in EPRI NP-3967 (Reference 10). The beta factors were applied to systems at SQN in such a way that negated much of the redundancy and flexibility designed into these systems. Beta factors cannot accurately be applied without a detailed review of common failure experience to determine which failure modes are applicable to a specific situation and how much equipment would be affected by particular common failure mode. The assumption that failure of additional redundant components is guaranteed after failure of the second is too severe without some justification based on plant-specific data analysis. Also, there is no justification to apply the same beta factors derived for standby equipment to normally operating equipment.

The application of beta factors in this analysis has a significant effect on system failure and should receive more detailed attention if accurate conclusions are to be drawn from the results. Sensitivity studies on beta factors represented the upper and lower bounds of the core damage frequency analysis. Also, these assumptions had a direct effect on the containment and consequence analyses because of the severe plant damage states dictated by these assumptions. Since beta factors are responsible for such a large fraction of the variance in latent fatalities, they merit careful attention.

2.5.5 Pessimistic Success Criteria

The NUREG-1150 core damage frequency analysis assumes all loss of power transients (T_1) and loss of DC power transients (T_{DCX}) result in opening the power-operated relief valves (PORVs). This is an extremely conservative assumption. WCAP-9600 (Reference 15), Table 4.5-2, shows the expected primary system pressure response for a loss of offsite power with, or without, control systems available. The PORV setpoint is not reached in either case. If auxiliary feedwater (AFW) is available, T_1 and T_{DCX} should not result in lifting the PORVs. Furthermore, since the PORVs fail closed on loss of power, the loss of DC power transients should not result in opening the PORVs. This assumption becomes significant because both PORVs are now required to reclose for T_1 and T_{DCX} transients.

The NUREG-1150 core damage frequency analysis assumes that both pressurizer PORVs are required for successful bleed and feed. Although it is preferable to have both PORVs held open, the analyses TVA has performed as part of the IDCOR program suggest the opening of only one PORV for heat removal would not result in core damage. Also, analyses documented in EPRI NP-3835 (Reference 16) show that one PORV is sufficient to prevent core damage.

No credit is given for the operation of the AFW system during small LOCAs. Although the AFW is not capable of preventing core damage indefinitely, it is able to significantly delay core uncovery which should be significant if recovery actions were incorporated into the event analysis. Therefore, the operation of AFW during small LOCAs results in a less severe plant damage state even if core damage is not avoided since the timing to core degradation is extended.

2.5.6 Containment/Consequence Analysis

NUREG-1150 utilized limited plant-specific containment analyses. This in combination with conservatisms noted in section 2.2 has resulted in an unrealistic portrayal of SQN. The lack of realistic detailed analysis of the dominant core damage sequences allowed the propagation of conservative assumptions into the containment consequence analysis. This results in an unrealistically high frequency of severe plant damage states.

2.5.7 Adequacy of Event Systems Analysis

Although the systems analysis appeared fairly crude, detailed comments will be withheld pending publication of the system fault trees and the data used for quantification.

The event analysis lacked a detailed assessment of the realistic plant response to the most important sequences. These sequences should have been evaluated before being used as input to the containment and consequence analysis. This lack of realistic analysis invalidates the risk reduction analysis since many of the proposed options would not result in changes to the system models if the assumptions used in the base case were also applied to the risk reduction analysis.

2.6 Uncertainty Analysis

Specific applications of the uncertainty analysis have led to perplexing results. The frequency calculated for T_{CCW}, is 2.7×10^{-5} per year if propagation of mean values is used, but the plant damage state which corresponds to this squence has a mean frequency of 4.9×10^{-5} per year. Similarly, the total core damage frequency for the dominant sequences calculated from a propagation of mean values is 8.6×10^{-5} per year, but the calculated mean frequency is 1.0×10^{-4} per year.

NUREG-1150, Appendix K, The Approach to Evaluation of Uncertainties, has extensive discussions of various ways to treat multiple statistically dependent failures. However, the NUREG-1150 analyses assume that failure of redundant equipment beyond the second component (third, fourth, etc.) is guaranteed if the second component fails. This assumption is not clearly stated in Appendix K. This indicates that the uncertainty analysis may be unfairly biased.

It is believed that the uncertainty issue within the NUREG-1150 analysis is one of the most controversial. In the early stages of the analysis, the optimistic-central-pessimistic (OCP) concept was utilized in the quantification of the containment event trees.

The OCP method consists of utilizing expert opinion to obtain branch point probabilities for the various top events (e.g., hydrogen burn at time of vessel breach). The pessimistic probability leads to high probabilities being assigned to pathways through the event tree which result in high source terms and low probabilities to event pathways resulting in low source terms. Conversely, the optimistic approach results in predictions being opposite that of the pessimistic approach. The central approach, therefore, lies somewhere between the optimistic and pessimistic approach. Furthermore, no apparent weighting factors are utilized in this approach.

This approach led to much controversy during the review (NUREG/CR-4569, Reference 17) of the containment event analyses (NUREG/CR-4700, Reference 18). The use of the OCP method led to statements by one of the reviewers that the NUREG-1150 results were pessimistically biased while one of the authors replied that he viewed the reviewer's comments as an "espousal of the optimistic" viewpoint.

In the latter stages of the analysis (see page D-41 of Appendix D of Reference 1), the Limited Latin Hypercube (LLH) analysis was used. This LLH technique uses statistical sampling to evaluate the relationships between various issues (e.g., hydrogen combustion) and risk. "The range of values attributable to each of the issues considered and the relative degree of belief of values within the range were provided by a team of experts in the appropriate technical area." This technique has not been given the peer review that the OCP method has received because it was used in the later stages of NUREG-1150 development.

One of the real dangers, and it is obvious upon reviewing the NUREG-1150 support documents, is the incorporation of expert opinion beyond the knowledge and expertise of the experts themselves. Not only were the experts asked to give opinion on very diverse and complex phenomenologies, but were then asked to apply these decisions to various containment designs which have very unique and complex behaviors to the various phenomena. Therefore, at best, the use of expert opinion could lead to fairly limited confidence in the overall behavior of containment systems to these phenomena.

It should be pointed out that the risk reported in the IDCOR analyses lies below the risk ranges reported in NUREG-1150. If a truly objective analysis was performed, it is not clear why the lower bound on risk reported by NUREG-1150 was not extended to include the IDCOR results.

If certain IDCOR arguments are dismissed, then the reasons should be stated and documentation provided. It appears that the use of expert opinion is unavoidable; however, these opinions should be documented as carefully and completely as possible for the benefit of experts with differing opinions. Presently, the documentation in this area of NUREG-1150 is insufficient and unacceptable.

Referring to Reference 1, Appendix D, page D-43, it was stated that the LLH analysis showed that the conditional probability of early containment failure is closely tied to the magnitude of a hydrogen burn occurring at the time of reactor vessel breach. The issues of hydrogen burns at vessel breach and the magnitude of radionuclide releases generated during core-concrete interactions are also shown to be important contributors to the variance in early fatalities.

However, a surprising result of the LLH analysis is that the variance for latent cancer fatalities is associated with assessing common-cause failures in the accident sequence analysis (beta factors). This shows that the coupling between the systems analysis with the containment response, source term, and consequence analysis may be strong and that common-cause failures should be examined carefully. However, this result is based on the NUREG-1150 finding that the loss of CCS sequence is a significant contributor to the SQN core damage frequency.

In contrast, the preliminary findings of the SQN IPE, Reference 8, show that the loss of CCS disagrees with the NUREG-1150 assumption that a loss of CCS will result in failure of the centrifugal charging pumps to deliver flow to the RCP seals (see section 2.5 for additional discussion).

The variance of latent cancer fatalities may not be influenced by common-cause failures as much as NUREG-1150 suggests. Therefore, based on the preliminary SQN IPE findings, this conclusion may be incorrect and require reexamination.

2.7 Operator Actions

The modeling of operator actions in the NUREG-1150 analyses was over-conservative. This is particularly evident given the fact that numerous procedures and alternate equipment alignment could be initiated in order to minimize the effects of severe accident sequences.

A consistent recovery analysis is not found in NUREG-1150. Most systems common to both units are either artificially separated in the analysis or cross connections are excluded by the analysis assumptions. In addition, some initiators such as loss of component cooling water and loss of DC power are assumed non-recoverable even though there are several options available to the plant operators.

2.8 Risk Estimation

Referring to Figure 7.10 of Reference 1, Volume 1, the risk estimation is indicated for SQN. The risk for early fatalities is estimated between 4×10^{-6} and 3×10^{-4} per year. In contrast, the IDCOR analysis predicts health effects significantly lower than those reported in NUREG-1150. A number of factors as discussed in section 2.2 contribute to this fact.

It should be noted that the latent fatality and population dose risk estimated by IDCOR is below the range of risks estimated in NUREG-1150.

2.9 Calculation of Risk Reduction

The calculation of the effects of various preventative and mitigative options have on risk reduction is at best misleading. The risk reduction analysis assumes that: (a) "each option functions properly as designed," (b) "the option under consideration was designed, installed, and operated properly," and (c) "did not interfere adversely with other plant features in a subtle and unknown way." The combination of these assumptions and limitations of the NUREG-1150 analysis resulted in an analysis which for several options has apparently calculated the effect of changing the NUREG-1150 analysis assumptions rather than the effect of plant modifications.

For example:

Option P3 - Improved DC power system. This option evaluates upgrading administrative controls to ensure availability of the fifth battery set and altering testing and maintenance practices to minimize the possibility of sources of failure shared with other battery trains. Since neither the administrative controls nor the test and maintenance practices were evaluated for the baseline analysis, the evaluation of the proposed improvements has no realistic significance.

Option P4 - Increased duration of injection phase for small LOCAs. This option evaluates adding steps to the procedures to instruct the operators to secure the spray system under certain conditions. The procedure which addresses small LOCA response already includes steps which instruct the operator to place the containment spray pumps in standby if containment pressure is reduced below the initiation setpoint (2.81 psid). However, the NUREG-1150 baseline analysis did not evaluate specific operating procedures in the analysis and, therefore, the effect of this procedure was not included in the baseline analysis. Therefore, the evaluation of this "improvement" has no realistic significance.

Option P5 - Improvement of the CCS. This option proposed more stringent controls on the number of CCS pumps that could be taken out of service for maintenance at any given time. However, maintenance procedures were not addressed in NUREG-1150 and maintenance unavailability was not a factor in the evaluation of the loss of CCS. The assignment of a generic beta factor to the operating pumps was the dominant feature in the calculation of CCS unavailability. The only factor that would have a significant effect on the calculation of component cooling availability is the determination of the probability of common cause failures. The proposed "improvement" has no realistic significance.

Option P6 - Upgrade of the AFW system. This option examined improvement of the AFW system in installing cross-connections in the discharge lines between the two SQN units. This option failed to consider the possibility of creating a common failure mechanism, such as steam binding, for AFW at both units.

The effects of options P2, P3, P5, and P6 should be collectively known as an option to fix generic beta factors since they are all significantly affected by the effect the proposed option has in the calculation of common cause failure. Additional detail should be provided on the calculation of the worth of the various options. This information should include a basic description of the option, how the improvement in reliability is assessed to that option, and how this was transformed into risk reduction.

SUMMARY

The configuration and design of SQN satisfies the current NRC safety goals but may not satisfy future NRC safety goals if NUREG-1150 is used in its present condition as the decisionmaking tool. The recent SQN IPE and the previous IDCOR studies produced results and conclusions that differ from those of the NUREG-1150 study. Although the uncertainty band in NUREG-1150 is large, the risks predicted by IDCOR still reside below the lower bound of the NUREG-1150 range. This implies that the views of the industry were not incorporated adequately in the NUREG-1150 study, thus leading one to conclude that the uncertainty analysis in the NRC study is a major issue.

The phenomena which appears to have the greatest impact on early containment failure and early fatalities is hydrogen generation and combustion. The IDCOR calculations and modeling assumptions differ greatly from the NUREG-1150 analysis. These differences have been documented and the basis for the phenomenological modeling assumptions compared to experimental evidence in order to provide validation. In addition, the assumptions concerning the fission product releases during the corium-concrete interactions may have a strong impact on the predictions of early fatalities. Again, these differences have been identified in the past with IDCOR providing a basis for the modeling assumptions and experimental validation where possible.

Although very important, the issues of direct heating and steam generator tube failure have been considered secondary issues for this review. However, as time permits, these issues will be given more emphasis for comparison to IDCOR calculations and modeling assumptions.

The majority of the issues that drive the overall risk estimates for the reference plants are related to those issues concerning the phenomena that control early containment failure. However, certain systems' model assumptions also had an impact on the dominant sequences that led to core damage. These assumptions include those of initiating event frequencies, common cause failures, success criteria, recovery actions, RCP seal LOCA model, and dependence of centrifugal charging pumps on CCS.

The SQN IPE identified and documented several areas where the NUREG-1150 analysis assumptions were either outdated or incorrect. The SQN IPE provided a tool for identifying and applying the most recent plant-specific modeling assumptions to determine the basis for over-conservatisms used in the NUREG-1150 analysis. It is recommended that NRC utilize this information to correct the technical deficiencies in the current draft of NUREG-1150.

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