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NS-EPR-2732

March 16, 1983

Mr. M. W. Jankowski
Fuel Behavior Branch
Division of Accident Evaluation
U.S. Nuclear Regulatory Commission
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Rockville, Maryland 20850

Dear Mr. Jankowski:

We have reviewed the NRC document, "Radionuclide Release Under LWR Accident Conditions," NUREG-0956, Draft, as requested by your letter of January 14, 1983. Generally, the document represents a useful update of the fission product transport and deposition estimates for core melt accidents with respect to those presented in the earlier NRC Report, "Technical Basis for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, 1982. However, the draft document reveals many important omissions as a result either of calculations that were not completed at the time the draft was issued, or failure to consider some of the important deposition steps. We urge that these omissions be treated and appropriate estimates for them be included in the next draft of the document.

Generally, the document considers the potential for radionuclide retention during the two principal transport steps, transport of released radionuclides from the core region through the reactor control system, and transport and deposition in the containment. Westinghouse has recently produced estimates of fission product retention during transport in conjunction with UKAEA as part of the estimates of risk from core melt accidents for the Sizewell-B plant (proposed for construction by the Central Electricity Generating Board). These estimates are reported in the UKAEA document, SRD-R256, November, 1982. Generally, estimates of fission product retention are similar to those in NUREG-0956 for sequences where the same transport steps are considered (note that the results in SRD-R256 are reported in a different format than those in NUREG-0956). The results in SRD-R256 show that there is a significant dependence of the magnitude of reactor system deposition on the location of breaks in piping systems (hot leg or cold leg). Results in NUREG-0956 also show this dependence even though such a conclusion is not drawn. For example, the large hot leg break sequence (AB) shows little retention while there is substantial deposition for the small cold leg break (S) sequence. Parallel consideration of both hot and cold leg break retention for large break (A), small break (S), and containment bypass (V) sequences is needed in NUREG-0956.

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With regard to the TMLB' sequence, the sequence does not appear to have been treated in enough detail. Sweeping of the vessel controls by steam when the core debris material slumps into the water in the bottom of the vessel needs to be considered with concurrent potential for deposition within the reactor system. The assumption of essentially instantaneous vessel melt-through at core slumping is unduly conservative. Potential for fission product retention in the relief line downstream of the pressure relief valves and in the quench tank also needs to be estimated (see SRD-R-256).

With respect to containment transport and deposition, the improved aerosol models employed in NUREG-0956 show retention in the containment to be markedly greater than earlier predictions with the CORRAL code for sequences in which containment failures does not occur for a number of hours following fission product release to the containment. These more realistic calculations showing greater retention are encouraging.

Large releases to the environment are still predicted in NUREG-0956 for sequences in which containment failure occurs soon after fission product release to the containment. However, recent studies for large dry PWR containments in which Westinghouse has participated (Zion, Indian Point, Sizewell-B), all show the probability of early containment failure to be markedly lower than those for delayed failure. We urge that the NRC report provide the needed perspective with respect to the relative magnitude for early and delayed containment failures so that the significant role of the containment in reducing radionuclide release to the environment and minimizing risk will be properly recognized.

The role of the auxiliary building or other secondary containment structures in reducing radionuclide release to the environment has not been adequately treated in NUREG-0956. The treatment of auxiliary building retention for the V-sequence in NUREG-0956 is brief, difficult to follow and the assumed building parameters do not appear to be proper. Potential for holdup and retention in secondary structures also needs to be treated for the failure to isolate containment sequences (β failure mode). We urge NRC give needed attention to this important radionuclide retention step.

In summary, NUREG-0956 provides a useful update to earlier NRC estimates of fission product retention for core melt accidents. However, significant additional work is needed to place the limited calculations in perspective. We urge a careful review of the recent radionuclide transport study done for Sizewell-B. While the estimates in that study are not based on an extensive set of computer calculations, they do represent a more complete consideration of the effects of overall system, containment, and secondary structure configurations, and of break locations on the potential for radionuclide retention during radionuclide transport following a core melt accident than presented in NUREG-0956.

Attached are specific comments developed during the review of NUREG-0956. The comments transmitted include those of both our assigned peer reviewer, Dr. D. H. Walker, and of the Westinghouse observer at the peer review meeting, G. T. Rymer.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

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cc: M. R. Hayns - UKAEA

NUREG-0956 COMMENTS

SPECIFIC COMMENTS

p.3-1, last paragraph -

Recent risk studies have shown that the Surrey plant, utilized as the PWR for analysis in the RSS(1), is not typical of current generation PWR's, with large dry containments, in a number of ways important to the radionuclide containment function including design and reliability of containment heat removal systems and pressure containment capability(2,3). While the desire to retain Surrey in these analyses for comparison to the RSS results is commendable, similar analyses for a current generation large dry PWR should also be performed.

p.3-3, second full paragraph -

Current risk studies have shown uncertainties in the magnitude of the radionuclide release to the environment for core melt accidents is one of the major uncertainties associated with overall estimates of risk for a nuclear power station(2,3,5). Certainly some estimate of source term uncertainty is an essential next step to permit meaningful application of the NUREG-0956 results.

p.3-5 and 3-6, list of uncertainties

Item (1) - need to indicate which way could the results be biased?

Item (6) - integral experiments to validate primary system transport models do not appear to be forthcoming, unless perhaps LOFT is used for this purpose. What does NRC believe is the implication of there likely being no experimental validation of the primary system transport models?

Item (10) - TRAP-MELT parametric calculations to determine the importance of deposition velocities to overall release to the environment for important sequences are needed. If the effect on release is large (orders of magnitude), TRAP-MELT calculations are not very useful or meaningful until deposition velocities are determined.

p.4-10, last paragraph -

The degree of retention in the auxiliary building should also be considered for the V-sequence.

p.5-12, 1st paragraph -

Statements at the end of the paragraph need to be softened. For some PWR's, V-sequences are dominant contributors to risk. For such sequences, source material released ex-vessel is not efficiently transported to the environment. Also for

some PWR designs, the ex-vessel debris will likely be cooled and certainly covered by water. For such designs, release to the environment may not be dominated by fission product released after vessel melt-through even for delayed containment failure sequences.

Section 7.2.2 Deposition for TMLB' -

1. Studies done for the Sizewell-B PWR indicated steam generation, as the hot core debris migrates to the lower plenum, is sufficient to sweep the radionuclide inventory from the vessel, into the pressurizer and out the pressurizer relief valve before vessel failure (see reference 4). Judgment would also lead one to believe that migration of core debris material into the water pool in the lower head would occur over a period of time such that sufficient steam would be generated and time available (of the order of 10 minutes or more) for sweeping of the pressure vessel and pressurizer of released fission products and aerosols prior to vessel melt-through.

2. For such a sequence, radionuclide deposition in the relief line between the relief valves and the quench tank needs to be considered. Deposition could be substantial (see reference 4).

3. Deposition in the quench tank should also be estimated for the TMLB' sequence both for cases where water is still present in the lines and for cases where the tank is emptied (tank surface temperatures are likely to be cool).

V-Sequence, Tables 7-11 and 7-12 -

Tabulated reactor system fission product retention values are for the cold leg (discharge) side V-sequence which was the type of sequence considered in the RSS. With the system changes incorporated in most designs to enhance check valve testing, hot leg (suction) side failures exhibit higher frequencies for many current generation PWR's (see references 2 and 3). For such sequences, the steam generators, with their cool surfaces and potential for radionuclide retention, are not part of the flow path and hence predicted reactor coolant system retention is reduced.

Section 7.3.2, TMLB' sequence -

This section discussed the containment failure modes, they being an early failure at 89 psi (γ) and a delayed failure mode (42 hours) at 100 psi (δ failure mode).

The phenomena producing the 89 psi pressure spike is not identified. If the pressure spike is due to hydrogen burn, the occurrence of the spike is unlikely since concurrent release of steam and hydrogen in this sequence is likely with resultant steam inerting of the containment. Further, the probability of failure at 89 psi is low. The report should clearly identify that the likelihood of the γ failure mode is much less than that of the δ failure mode for this sequence.

Section 7.3.4, V-sequence, 1st paragraph -

This section indicates that vaporization release will be treated in more detail in the next set of calculations. When this is done retention of the vaporization source within containment should also be considered since this source material will likely mix with the containment volume at large and be deposited rather than immediately migrating back through the melt hole in the reactor vessel and being transported to the break.

EDITORIAL COMMENTS

p. 6-33 -

Meaning of entries under slab at bottom of page are unclear.

p. 6-36 -

Were H1M and H10 intended to be the same ?

Equation for σ (tensile stress of vessel) is unclear.

p. 6-37 -

For entries (2) and (3), Surrey is not typical of current large dry containment PWR's.

p. 6-42 -

Quoted volume for auxiliary building seems too small by about a decade. Quoted areas also seem low. Was only part of a building or a compartment considered?

p. 7-26 and 7-27 -

Containment leak rates of 1%/day are used in the comparative calculations. Design basis leak rates in the range 0.1%/day or more are typical of large dry containments.

p. 7-28, 2nd paragraph -

Words should reflect the need to compare Tables 7-16 and 7-17. Generally writeup on this page needs editing.

GENERAL

1. Recommend on temperature system, °C or °F, be selected and used throughout the report.

REFERENCES

1. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, WASH-1400, (October 1975).

2. "Zion Probabilistic Safety Study," copywrited by Commonwealth Edison Company, Chicago, (1981).

3. Indian Point Probabilistic Safety Sutdy, copywrited by Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., (1982).

4. M. R. Hayns, F. Abbey, P. N. Clough, I. H. Dunbar, and D. H. Walker, "The Technical Basis of 'Spectral Source Terms' for Assessing Uncertainties in Fission Product Release During Accidents in PWR's with Special Reference to Sizewell-B," SRD-R256, UKAEA, November, 1982.

5. J. T. Larkins and M. A. Cunningham, "Nuclear Power Plant Severe Accident Research Plan," NUREG-0900, V83, p. 5-55, U.S. Nuclear Regulatory Commission.