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February 24, 2003

MEMORANDUM TO: Roy Zimmerman, Director
Office of Nuclear Security and Incident Response

FROM: Ashok Thadani, Director
Office of Nuclear Regulatory Research

SUBJECT: PRELIMINARY VULNERABILITY ASSESSMENT RESULTS (U)

(U) Attached is a brief report which provides insights and preliminary results from analyses of spent fuel pool events performed as part of ongoing work in the integrated vulnerability assessment. While some of these results are very recent, we provide them so that you may be apprised of our best current understanding. These analyses have been performed using state-of-the-art methods and treat the scenario in a consistent integrated manner. We believe these analyses are a significant improvement over past predictions of accident response. RES plans to continually update you as new analyses are completed. In the long term, we plan to issue a report on the consequences of spent fuel pool accidents which will supplant the analyses in NUREG-1738, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, February 2001.

(U) These analyses were performed to address the fully and partially drained pool configurations for an operating reactor and have addressed, where applicable, comparisons with the generic study performed in NUREG-1738. Calculations address the decay time required for air cooling, effect of fuel location in the pool, air oxidation, pool boiloff and transient drain down time and minimum water level for steam cooling. In general, the analyses show significant departure from the generic study and indicate that spent fuel is much more easily cooled in a fully drained configuration. The partially drained configuration still appears to be limiting for older fuel but this configuration needs further analysis to address all heat transfer mechanisms.

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~~Daniel H. Dorman, BC, RES/DET/ERAB~~
~~Name, Title, Organization, Date~~ 2/24/03

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(U)

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In NUREG-1738 all of the fuel was assumed to release a significant fraction of its fission product inventory. The timing of the release predicted in our preliminary analysis is much less severe (starts later and extends over longer interval) than assumed in NUREG-1738. This reduction in the radionuclide release alone has significant impacts on offsite consequences.

Ex. 2

By contrast, NUREG-1738 assumed no mitigation by the building or its remnants.

(U) In this report, we have also provided some initial insights on fuel loading and makeup sources of water and we expect further work to provide more of the same along with greater clarification and confirmation of these points.

(U) The analyses described herein represent a substantial improvement in the way we perform evaluations of spent fuel pool accidents. More work is essential to confirm and expand these analyses but this preliminary evaluation is providing a strong base for our ongoing vulnerability assessment. We will provide you with additional information as we proceed.

Attachment: As stated

cc:
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PRELIMINARY VULNERABILITY ASSESSMENT RESULTS (U)

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Attachment

PRELIMINARY VULNERABILITY ASSESSMENT RESULTS (U)

SUMMARY(U)

(U) As part of the integrated vulnerability assessment performed by RES, we are performing revised analysis of accident progression, fuel damage and offsite consequences using state of the art methods and best estimate approaches with consideration of uncertainty. We are incorporating the results of severe accident research insights, including new computational techniques and reassessing assumptions that have commonly been made in past analyses. The analyses are being performed for both reactors and spent fuel pools. This brief paper provides a summary of the insights of the ongoing work to date, specifically in the area of spent fuel pool analyses.

(U) Preliminary analyses of spent fuel pool events suggest that these more realistic treatments of heat transfer mechanisms and integrated treatments of atmosphere composition and fluid flow produce more favorable predictions (compared to some past predictions including NUREG-1738) of fuel coolability and fuel degradation when coolability is not assured. NUREG-1738 calculations indicated that about 4-5 years of decay is needed before air cooling, for a fully drained pool, is sufficient to preclude a zirconium fire. Our calculations suggest fuel is air coolable, for a fully drained pool, at much earlier times.

Ex. 2

(U) For cases where fuel was not air coolable, our preliminary analyses also show more margin to fission product release, i.e., time available for corrective action, from previous calculations. Ex. 2

Ex. 2

SGI Ex 3

SGI Ex 3

Ex. 2

(U) The first preliminary integral calculations of a spent fuel pool event have been very recently completed. These are the first calculations of their type we know to be performed. The assumed boundary condition was The integral calculations show a very large reduction in the magnitude and rate of the radiological release to the environment from that assumed in the generic NUREG-1738 analyses and earlier analyses. Ex. 2

(U) The preliminary integral analvses, for the postulated scenario, indicates that NUREG-1738 assumed all of the fuel released its fission products at the same amount. Ex. 2 Additionally, our preliminary integral analysis In NUREG-1738, it was assumed that the initiating event, a large earthquake, destroyed the reactor building such that it no longer served as a hold -up volume and there was no specific deposition. Ex. 2

Ex. 2

(U) In the preliminary integral analyses the total offsite releases (expressed as a release fraction of total inventory) from the NUREG-1738 assumptions (based on cesium release as a metric). The calculated timing of the release is also significantly different, i.e., less severe, from the assumed timing in the NUREG- 1738 consequence estimates. Releases are predicted to begin than assumed in the earlier generic calculations. The new calculated time interval available for evacuation of the EPZ Ex. 2

Ex. 2

(U) Future work is expected to provide insights as to whether could provide even greater benefits for coolability and prevention of fuel damage. Ex 2

(U) The preliminary analyses have provided numerous insights and an improved understanding of the more realistic range of fuel damage and offsite consequences resulting from terrorist attacks. We have identified some of the insights on preventing and mitigating the consequences of spent fuel pool events. These calculations need to be expanded, reviewed and confirmed (including experimental confirmation) and modified to include a consideration of uncertainty. Ex 2
Nonetheless, we think this ongoing work with its preliminary results, along with improved analyses of reactor events and alternative consequence modeling, is already providing a vastly stronger foundation for our vulnerability assessment.

PRELIMINARY VULNERABILITY ASSESSMENT RESULTS (U)

1. BACKGROUND AND OBJECTIVES (U)

(U) In response to the terrorist attacks on September 11, 2001, the NRC has undertaken an assessment of the vulnerabilities of nuclear facilities to a variety of threats. Evaluation of the vulnerabilities and the potential consequences of attacks on facilities guides the agency in assigning priorities, identifying possible compensatory or mitigating actions and ensuring security is appropriately addressed for both operating facilities and future designs. As a part of the agency's activities, RES has underway a detailed study of the vulnerabilities posed by attacks on operating nuclear power plants. This study addresses a range of threats including aircraft attack, vehicle bombs, cyber terrorism, and armed attack by intruders with possible assistance by an insider. The RES assessment of the operating plants includes characterization of the vulnerability of the reactor and the spent fuel stored on-site in the spent fuel pool. Vulnerability is being evaluated as an integrated risk assessment; a probability of occurrence is attached to the individual elements of the analysis and scenario, those events leading to damage of structures, systems, fuel damage, offsite release of radionuclides and public health consequences and environmental impacts. Ex. 2

(U) The objective of this brief paper is to summarize the results of preliminary analysis performed to date with regard to the prediction of accident progression and offsite consequences. Further, this paper focuses on the results of one area of analysis:

> Analysis of the phenomena associated with cooling, heatup, and possible degradation of fuel in the spent fuel pool together with potential offsite consequences in the event fuel heating is sufficient to result in fuel failure and significant radionuclide release.

(U) With respect to the analysis of spent fuel pool heatup, these analyses done as part of the vulnerability assessment have implications for the evaluation of other non-vulnerability related scenarios, such as those considered in NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants." The methodology developed for this vulnerability assessment is a substantial improvement upon the methods adopted for the generic decommissioning study and it is expected that future analyses of spent fuel pool accidents, regardless of the accident initiator, would benefit from the more mechanistic and realistic assessment of fuel thermal response, zircalloy oxidation and possible radionuclide release. Comparison of the ongoing analyses with the analysis in NUREG-1738 is provided.

rel

2. SPENT FUEL POOL ANALYSIS (U)

(U) The general methodology being used to evaluate spent fuel pool cooling under conditions where water has drained from the pool is based on a two tiered approach to evaluate the thermal hydraulic and natural circulation conditions. In order to evaluate details of natural circulation flow through the fuel assemblies and rack cell structures, a CFD code is being used because of its inherent capabilities to simulate multidimensional flow, detailed flow losses through a complex geometry and free shear flow conditions associated with buoyant plumes. The CFD code is also being used to assess mixing of the plume exiting the pool with the bulk building atmosphere and the boundary condition associated with the airflow to the pool. To simulate the overall system response we use the MELCOR severe accident analysis code which includes modeling of the transient draindown, boiloff, natural circulation, zircalloy oxidation, material heatup, melting and relocation as well as potential fission release and transport. The MELCOR code analysis, using the conventional control volume code approach, will use the insights from the detailed CFD model to benchmark its basic thermal hydraulic model of volumes, flow paths and resistances.

(U) To further guide the development of the MELCOR integral spent fuel pool model, we have also devised an approach wherein we construct a simple separate effects model, as shown in Figure 1, simulating 5 fuel assemblies, with a central assembly and its 4 adjoining neighbors. With this separate effects model we can investigate basic phenomena of air and steam cooling, impact of flow resistances on natural circulation cooling and issues attendant to the failure and melting of the cell walls (boral or boraflex dividers). The first set of calculations were done with a model simulating details of the Fort Calhoun (CE PWR) spent fuel pool assemblies and cell racks (existing detailed information was readily available). This racking is a high density Holtec design, generally representative of numerous domestic BWR and PWR plants as well as foreign reactors.

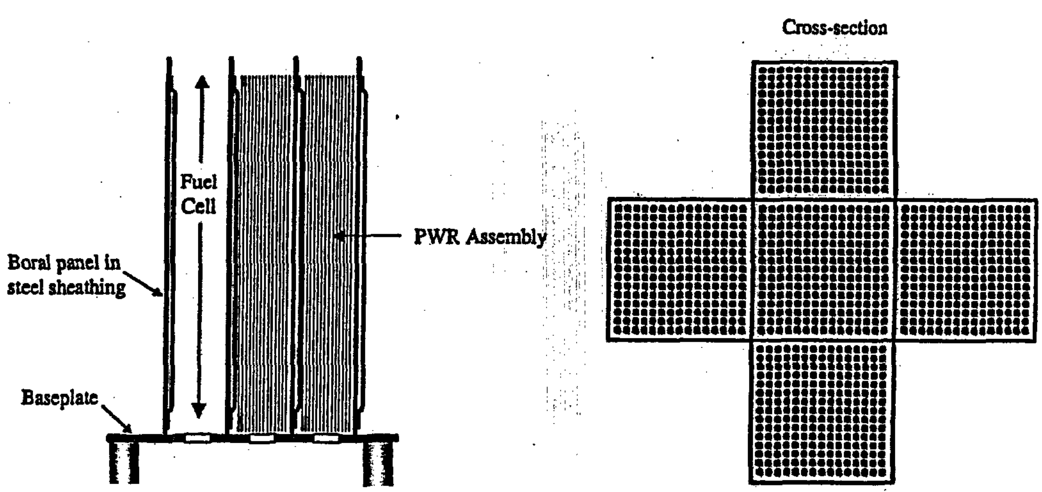

Figure 1. Separate Effects Model (U)

(U) The first calculations were performed to test the model and assess the effectiveness of air cooling for various decay times.

Ex 2

The results of air cooling calculations, assuming the water has drained from the pool are shown in Figure 2.

Ex 2

(U) The curves in Figure 2 show the thermal response of fuel cladding for

The figure indicates, for the separate effects calculation, that fuel which has could be air cooled; its temperature would rise above the initial temperature but would stabilize and reach a new equilibrium at a temperature below that at which zircalloy oxidation would cause an uncontrollable increase (i.e., "zirc fire"). Other

 Ex 2 portions

calculations indicated (though not shown in the figure) that the

Fuel failure could be expected to occur at a temperature of and fission products would begin to be released but significant releases would not occur until somewhat later. The sharp upward temperature ramp, is due to the exothermic energy addition associated with cladding oxidation.

(U) Because natural circulation cooling with air is influenced by the resistance (e.g., form and wall losses) of the flow path we looked at potential locations where the localized resistances of the path may be more severe. Shown in Figure 3

Ex 2 portions

(U) Because the fuel assemblies located over rack feet have a much larger flow path resistance, the decay time required for those assemblies to be air coolable increases dramatically.

This can be compared to the coolability for other assemblies shown in Figure 2 and described above. This conclusion one could draw from this analysis is that

although we recognize that other design constraints may exist. There is a point here relevant to the NUREG-1738 study concerning flow blockage, in that flow blockage was cited as a reason for concluding that in some instances fuel may never be sufficiently decayed to be coolable by air.

(U) Additional analyses, not described in detail here, have also been performed to examine the extent to which the limiting of oxidant (air starvation) was reducing the rate of cladding oxidation. MELCOR modeling accounts for the composition of the atmosphere (oxygen, nitrogen, and steam) and thus includes the effects of oxygen deprivation.

Future analyses will clarify this effect on fuel heatup.

Ex 2 portions

Ex 2

was recognized in the NUREG-1738 study, which then addressed the subsequent heatup with an adiabatic heatup estimate.

(U) Even in the case of

there may be substantial time for corrective action to restore the water inventory or cooling of the fuel. A series of simple calculations were performed, in this case simulating a complete pool, (no longer using the separate effects model), to assess the time for the water to drop to various key elevations. These calculations were done roughly simulating the inventory and decay heat level in the

Ex 2 portions

An observation can be made relative to this pool based on our existing information(which is ~~being~~ confirmed). Unlike the configuration assumed for pools in decommissioning reactors,

~~(SGI)~~ The level calculations were performed for various hole sizes

Ex 2 + Ex 3 portions

Ex3

(U) Discussions with local firefighting experts confirmed that, as an example, pumper trucks commonly used in the Baltimore-Washington area have a rated capacity of 1000 -1500 gpm. (Some trucks have capability up to 2000 gpm). Losses (line losses and elevation changes) will reduce that capability. The ability to draw from a distant source (e.g., pond, river, or tank) and deliver flow may be limited on the suction side by the fixed piping carried or available but discharge over relatively long distances is feasible. Trucks can also be aligned Ex 2

(U) A part of the examination of spent fuel pool coolability and the potential for fuel damage (and concomitant radionuclide release) is the integral analysis Simulation is necessary to capture the potential for fission product mitigation, by flooding of adjoining rooms and to model the mixing and inlet air flow to the spent fuel pool. Ex 2 then significant fission product deposition may occur even though the fuel has released its volatile radionuclide inventory. This would, in turn, significantly reduce offsite consequences. Where the separate effects model was used to simulate details for a 5 assembly array of fuel, the integral model will simulate the 3000 some assemblies which comprise the pool inventory. Based on the general distribution of fuel assemblies a MELCOR model was constructed

(U) The integral model has as an objective, in addition to the calculation of the onset of fuel damage, the prediction of the extent to which propagation of fuel damage may occur given heatup of the highest decay heat assemblies. Since the decay heat levels may vary widely within the pool across major regions, it is plausible that relatively smaller regions, even if they reach high temperatures associated with rapid zirconium oxidation, may not heat adjoining large colder regions. At a minimum, it is reasonable to posit that the time for the fuel through out the pool to release its fission products may be influenced by the relative distribution of

Ex 2 + Ex 3 portions

power within the pool. In the NUREG-1738 study it was assumed, from a bounding standpoint, that all of the fuel released its fission product inventory simultaneously, regardless of power level, over an interval of 30 minutes. In NUREG-1738, it was also assumed that radionuclide release began in about 1 hour. The integral analysis will provide a much more realistic characterization of the timing of the radionuclide release together with a more realistic evaluation of the magnitude of that release.

(U) For the first preliminary analysis, it was assumed that

We have not as yet

The scenario chosen for this calculation is a draindown with air cooling. This type of scenario is most like those addressed in NUREG-1738 and is considered to be important for the vulnerability assessment. Unlike the NUREG-1738 study, in this analysis we further assume the drainage of water to occur not instantaneously but rather as a transient process

(U) Future calculations will explore the sensitivity to these analysis assumptions and address different scenarios,

Modeling issues related to

also need to be further addressed.

(U) The results of the preliminary integral analysis indicate differences from earlier spent fuel pool analyses. First, this analysis indicates that only the fuel

In NUREG-1738 and earlier analyses, it was assumed that the entire pool inventory released its fission products in the same amount, with no differentiation of the release fractions of radionuclides for the age of the fuel. This assumption was based on the premise that the zirc fire or fuel damage propagates through the entire pool. Table 2

Portions Ex 2

Source Term	Release Fractions								
	Xe	I	Cs	Te	Sr	Ba	Ru	La	Ce
NUREG-1738	1	.75	.75	.31	.12	.12	.75	.035	.035

*Assumes same fuel release fraction for ruthenium as for iodine and cesium.

Table 2. Offsite Release Fractions (U)

(U) The timing of the release of radionuclides from the damaged fuel in the spent fuel pool is also greatly changed from the NUREG-1738 assumptions in that it is less severe. The first offsite release of radionuclides occurs as compared to the assumed release start of 1 hour in NUREG-1738. In NUREG-1738, it was assumed that the release took place over an interval of 30 minutes, the preliminary analyses predicted the bulk of the release occurs

(U) Using the same as in NUREG-1738 and using the same , we performed analyses of the offsite consequences with the release fractions shown in Table 2 calculated from the MELCOR analysis of

Portions Ex 2