~~OFFICIAL USE ONLY~~

SANDIA REPORT

Revision 1
Draft

Evaluation of a BWR Spent Fuel Pool Accident Response to Loss-of-Coolant Inventory Scenarios Using MELCOR 1.8.5

Draft Completed: February 2004

Prepared by
KC Wagner*
R. O. Gauntt

Sandia National Laboratories
P.O. Box 5800
Albuquerque, NM 87185-0748

*Alion Science and Technology Corporation
6000 Uptown Boulevard, NE Suite 300
Albuquerque, NM 87110

Portions withheld - Ex 2 + Ex 5

[illegible]

6/17

~~OFFICIAL USE ONLY~~

EXECUTIVE SUMMARY

In 2001, United States Nuclear Regulatory Commission (NRC) staff performed an evaluation of the potential accident risk in a spent fuel pool (SFP) at decommissioning plants [NUREG-1738]. The study was prepared to provide a technical basis for decommissioning rulemaking for permanently shutdown nuclear power plants. The study described a modeling approach of a typical decommissioning plant with design assumptions and industry commitments; the thermal-hydraulic analyses performed to evaluate spent fuel stored in the spent fuel pool at decommissioning plants; the risk assessment of spent fuel pool accidents; the consequence calculations; and the implications for decommissioning regulatory requirements. It was known at the time that some of the assumptions in the accident progression in NUREG-1738 were conservative in order to simplify the analyses, especially the estimation of the timing and extent of fuel damage. Furthermore, the NRC desired to expand the study to include accidents in the spent fuel pools of operating power plants. Recognizing the simplifying conservatisms in the earlier analyses, the NRC has continued spent fuel pool accident research by applying best-estimate computer codes to predict the severe accident progression following various postulated accident initiators. This report presents a series of parametric calculations used to better understand the postulated accident behavior in a SFP.

The MELCOR 1.8.5 severe accident computer code [Gauntt] was used to simulate the SFP accident response. MELCOR includes fuel degradation models for BWR and PWR fuel, radiation, convection, and conduction heat transfer models, air and steam oxidation models, hydrogen burn models, two-phase thermal-hydraulic models, and fission product release and transport models. Hence, it contains the basic models to address phenomena expected during a spent fuel pool accident.

Ex. 5

Ex. 5

Ex. 5

Ex. 5

Table of Tables

Ex. 5

Ex. 5

Ex. 5

Table of Figures

Ex 5

Ex 5

Ex 6

Ex (

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

144

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Ex 5

Evaluation of a BWR Spent Fuel Pool Accident Response to Loss-of-Coolant Inventory Scenarios Using MELCOR 1.8.5

1. BACKGROUND

In 2001, NRC staff performed an evaluation of the potential accident risk in a spent fuel pool at decommissioning plants [NUREG-1738]. The study was prepared to provide a technical basis for decommissioning rulemaking for permanently shutdown nuclear power plants. The study described a modeling approach of a typical decommissioning plant with design assumptions and industry commitments; the thermal-hydraulic analyses performed to evaluate spent fuel stored in the spent fuel pool at decommissioning plants; the risk assessment of spent fuel pool accidents; the consequence calculations; and the implications for decommissioning regulatory requirements. It was known at the time that some of the assumptions in the accident progression in NUREG-1738 were conservative in order to simplify the analyses, especially the estimation of the timing and extent of fuel damage. Furthermore, the NRC desired to expand the study to include accidents in the spent fuel pools of operating power plants. Recognizing the simplifying conservatisms in the earlier analyses, the NRC has continued spent fuel pool accident research by applying best-estimate computer codes to predict the severe accident progression following various postulated accident initiators. The scope of the present analysis is to describe the response of a reference Boiling Water Reactor (BWR) SFP to both partial and complete-loss-of-coolant inventory accidents.

Ex. 5

The MELCOR 1.8.5 severe accident computer code [Gauntt] was used to simulate the SFP accident response. MELCOR includes fuel degradation models for BWR and PWR fuel, radiation, convection, and conduction heat transfer models, air and steam oxidation models, hydrogen burn models, two-phase thermal-hydraulic models, and fission product release and transport models. The code contains the basic models to address questions and phenomena expected during a spent fuel pool accident.

Ex 5

Consequently, MELCOR separate effect [Wagner, 2003] and computational fluid dynamics (CFD) [Chiffelle, 2003, Ross, 2003, and Suo-Anttila, 2003]

calculations were performed to develop a modeling approach for whole SFP severe accident calculations. These analyses helped guide the development of the whole SFP MELCOR model as well as help characterize many of the uncertainties and variability in the accident response.

Section 1.1 summarizes the two types of SFP accidents considered in the present report. They consist of a complete loss-of-coolant inventory and a partial loss-of-coolant inventory. Due to the varying phenomena, the calculated responses are discussed separately. Then Section 1.2 summarizes the structure of the report.

1.1 SFP Accident Scenarios

Ex 5

Hence, there are competing effects of the lack of a strong convective flow versus the benefits of some steam cooling and axial conduction to the water. In summary, the scenarios with water include (a) two-phase boiling, (b) an assembly flow rate that is strongly affected by the amount of boiling below the water surface, and (c) an assembly inlet temperature that is limited to the boiling point of water (i.e., the air cases are not similarly constrained).

The rate of oxidation of the zirconium cladding is the second key difference expected in a partial loss-of-coolant inventory accident. In particular, the fluid next to the zirconium is steam rather than air. Steam also reacts exothermically with zirconium but at a slower rate than oxygen. Furthermore, the byproduct of the steam-zirconium reaction is hydrogen. The hydrogen will replace the steam and retard or stop the zirconium/steam reaction. Consequently, the reaction would become "steam starved" and controlled by the rate of steam production by boiling below the pool level, which is expected to be very low for aged spent fuel. If there is adequate steam when the zirconium reaches high temperatures (i.e., >1500 K), then the power from oxidation reactions can be one or more orders of magnitude larger than decay heat. Therefore, there are two competing effects on the rate of fuel degradation relative to the complete loss-of-inventory accident scenario (i.e., as described in Section 1.1.1), (1) a lower, controlled oxidation effect (i.e., due to steam starvation) and (2) a much lower convective cooling rate (i.e., because the bottom of the racks are "plugged" with water).

Finally, a third new difference in the partial loss-of-coolant inventory accident is the behavior of the hydrogen. As hydrogen is produced during the fuel degradation, the hydrogen will rise and mix with oxygen in the air above the pool. Given the appropriate conditions, the hydrogen could ignite and possibly cause structural damage to the reactor building. Any damage or enhanced leakage caused by the pressurization from the hydrogen burn could increase the release of fission products and their associated adverse consequences.

1.2 Report Structure

In Section 2, a summary discussion is given on the reference BWR SFP. The modeling approach using MELCOR 1.8.5 is given in Section 3. Section 3 also includes a discussion of the decay heat modeling, the radial thermal coupling schemes, the reactor building model, the SFP nodalization, and the code model settings and sensitivity coefficients. The results of complete loss-of-coolant calculations are given in Section 4. The results from the partial loss-of-coolant cases are given in Section 5. An overall summary of their results is given in the Section 6. Finally, the references are provided in Section 7. Ex 5 Ex 5

2. SFP GEOMETRY

2.1 SFP Geometry

Ex 2

Ex. 2

Figure 2.1.1 Spent Fuel Pool Rack Layout.[4]

[4] All dimensions are in inches.

Table 2.1.1 Summary of SFP Attributes.

SFP Pool Characteristics	Description or Dimensions

Ex 2

Table 2.1.2 Spent Fuel Pool Rack Data.

SFP Rack Characteristics	Description or Dimensions

Ex. 2

2.2 SFP Assembly Geometry

Ex 2

Ex 5

Table 2.2.1 Fuel Assembly Data.

Assembly Characteristics	Description or Dimensions

Ex 2

3. ANALYSIS METHODOLOGY

The analysis methodology is described in the following subsections. Section 3.1 describes the decay heat calculations made for the various decay timings. The MELCOR SFP model is described in Section 3.2. Finally, Section 3.3 describes the various SFP calculations. Ex 5

3.1 Decay Heat Calculations

The SFP contains fuel assemblies from separate fuel offloads from the reference BWR reactor. Within a particular offload, variations in the assembly burnup and initial uranium loading are also present. The variations in these parameters lead to different levels of fission product decay heating. However, as the fuel ages, the heat produced by the fission product decay decreases. Ex. 2 Ex 5

Ex 5

7. REFERENCES

ANSI/ANS ANSI/ANS-5.1-1994, "Decay Heat Power in Light Water Reactors," ANS Standard, 1994.

Boyd, C., "Predictions of Spent Fuel Heatup after a Complete Loss of-Coolant Accident," NUREG-1726, July 2000.

Chiffelle, R., et al., "Analysis of Spent Fuel Pool Flow Patterns Using Computational Fluid Dynamics: Part 1 – Air Cases," Sandia National Laboratories, Informal Report, April 2003.

Collins, T. E., and Hubbard, G., "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February, 2001.

Gauntt, R. O., et al, "MELCOR Computer Code Manuals, Reference Manual, Version 1.8.5, Vol. 2, Rev. 2", Sandia National Laboratories, NUREG/CR-6119, SAND2000-2417/1.

Ross, K. W., et al., "Analysis of Spent Fuel Pool Flow Patterns Using Computational Fluid Dynamics: Part 2 – Partial Water Cases," Sandia National Laboratories, Informal Report, April 2003.

Suo-Anttila, A, J., Wagner, K. C., Gauntt, R. O., "Analysis of Spent Fuel Pool Flow Patterns Using Computational Fluid Dynamics: Supplement - Thermal Plume Response," Sandia National Laboratories, Informal Report, July 2003.

Wagner, K. C., et al., "Evaluation of Spent Fuel Pool Accident Response to a Complete Loss-of-Coolant Inventory Using MELCOR 1.8.5," Sandia National Laboratories, Informal Report, January 2003 [2003a].

Wagner, K. C., et al., "MELCOR 1.8.5 Separate Effect Analyses of Spent Fuel Pool Assembly Accident Response," Sandia National Laboratories, Informal Report, June 2003 [2003b].

Release