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CORE MELT PROGRESSION AND CONSEQUENCE ANALYSIS METHODOLOGY DEVELOPMENT IN SUPPORT OF THE SAVANNAH RIVER REACTOR PSA (U)

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A paper proposed for presentation at the
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INTRODUCTION

A three-level Probabilistic Safety Assessment (PSA) of production reactor operation has been underway since 1985 at the U. S. Department of Energy's Savannah River Site (SRS). The goals of this analysis are to: (1) analyze existing margins of safety provided by the heavy-water reactor (HWR) design challenged by postulated severe accidents; (2) compare measures of risk to the general public and onsite workers to guideline values, as well as to those posed by commercial reactor operation; and (3) develop the methodology and database necessary to prioritize improvements to engineering safety systems and components, operator training, and engineering projects that contribute significantly to improving plant safety. PSA technical staff from the Westinghouse Savannah River Company (WSRC) and Science Applications International Corporation (SAIC) have performed the assessment despite two obstacles: (1) a variable baseline plant configuration and power level; and (2) a lack of technically applicable code methodology to model the SRS reactor conditions. This paper discusses the detailed effort necessary to modify the requisite codes before accident analysis insights for the risk assessment were obtained.

Although the Level 1 PSA SRS methods for analysis of severe accident initiators and binning into plant damage states are similar to that available for commercial plants, the Level 2/3 portion of the analysis required code revision in virtually every aspect of the overall model. Modifications of key light-water reactor (LWR) codes had to be accomplished before integrated phenomenological assessments supporting an Accident Progression Event Tree could be finalized. Several accident and consequence analysis codes to predict primary system thermal hydraulics, fission product transport, confinement system thermal hydraulics, molten core-concrete interactions (MCCI), molten fuel coolant interactions (MFCI) and ex-plant consequence determination are discussed in this paper, along with associated model changes, the chief insights developed as a result of the analysis, and the extent of applicability elsewhere in the DOE complex.

METHODOLOGY

The modified computer codes necessary to support quantification of the in-plant Level 2 risk study are discussed initially. Next, revisions to the ex-plant consequence analysis model for onsite workers is summarized.

Phenomenology of postulated reactor accidents and the accompanying core melt progression is profoundly different in SRS reactors than in LWR counterparts. Most significantly, zircalloy-clad uranium dioxide fuel, high temperature-high pressure primary system conditions and containment parameters, will lead to fission product barrier challenges markedly different from those in a production reactor with aluminum alloy, uranium-aluminum fuel, low temperature-low pressure primary conditions operating with confinement. In the SRS PSA, integrated analysis tools were developed in three stages to provide temperature and pressure estimates, and later, fission product source terms, corresponding to severe accident scenarios. CONTEMPT4[1] was applied initially in a 1987 scoping analysis of confinement response. Heat loads were computed outside of the code and then incorporated in the analysis as tabular input to model core debris and structural sources of deposited energy. The second generation of analysis in 1989 - 1990 included the ability to model coupled thermal-hydraulic/aerosol/fission product behavior as part of CONTAIN/SR code. CONTAIN/SR was the primary tool used to support severe accident analysis in the 1990 Environmental Impact Statement for continued reactor operation at SRS and is tailored for fan/filter modeling based on CONTAIN 1.10.[2] By 1991, a more realistic filter compartment model and the ability to perform hundreds of sensitivity analyses to support quantification of the Level 2 APET led to adoption of the MELCOR/SR code as the primary tool supporting the SRS Level 2 PSA.[3] Additional demands for phenomenological insights due to MCCI and MFCI in SRS reactors required special-purpose models by SNL and the University of Wisconsin, respectively. The core debris-concrete interactions were estimated with a CORCON version that included aluminum and silicon species aerosols. The steam explosion tool allowed molten aluminum - coolant interactions to be predicted in terms of energetic yield, and steam and hydrogen over-pressures.

The MACCS analyses are conducted in a manner similar to the recent light-water reactor risk study[4]. However, the production reactor assessment is notably different in two key areas: (1) the presence of a large tritium source term component; (2) the requirement to estimate occupational risk to a large nearby workforce. The first difference is included in the MACCS analysis with the addition of a tenth fission product transport group as part of the source term with wet and dry deposition characteristics. The second difference required implementation of an modified MACCS model for modeling evacuation of the onsite worker population. The onsite analysis uses a model limiting evacuees to exit in a few preferential directions from the affected reactor . Thus the SRS PSA uses two MACCS models, one for calculation of consequences to the general public, and another for determination of the onsite worker consequences. Both calculations are based on MACCS 1.5.[5-7] Table 1 summarizes the major codes modified to support the integrated, phenomenological, and ex-plant consequence analysis for the SRS PSA.

SUMMARY

Differences between LWR and SRS production reactors demanded modification of methodology required to perform the Level 2/3 PSA at Savannah River. Ultimately, six computational tools were tailored to predict the in-plant response to, and the ex-plant consequences of hypothetical severe accidents. The resulting SRS PSA is based on the results obtained from the code enhancements. SNL supplied four of the methodologies, and was responsible for the model adequacy and functional acceptability review of three of the codes. Although the modified codes undergird the Level 2/3 PSA analysis at Savannah River, applicability throughout the DOE complex is broad.

Table 1.
Major Accident Analysis Codes Modified
Savannah River Site Reactor Probabilistic Safety Assessment (1985 - 1992)

Code	Authoring Institution/Assistance	Modification(s)
Integrated Analysis Codes		
CONTEMPT4	Brookhaven National Laboratory SAIC/Marietta	Intercompartment Flow Criteria; Heat Load Model Executed As Auxillary
CONTAIN/SR	Sandia National Laboratories WSRC & SNL	Limited Filter Compartment Model; Fan Model
MELCOR/SR	Sandia National Laboratories SAIC-Albuquerque	Detailed Filter Compartment Model; Fan Model
Phenomenological Analysis Codes		
CORCON	Sandia National Laboratory WSRC & SNL	Al and Si Chemistry Species Added; Decay Power Tables For U/Al Debris
Non-Equilib. Parametric Model	University of Wisconsin	Mixing, Propagation, Expansion Phase
Ex-Plant Consequence Analysis		
MACCS	Sandia National Laboratories	Evacuation Model For DOE Reservation Worker Population

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