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Validation status of design methods for predicting source terms

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ABSTRACT

Design methods have been developed internationally for predicting radionuclide source terms for modular HTGRs. These design methods have been used extensively to support reactor design and licensing. Numerous attempts have been made to validate these methods by comparing predictions with experimental data from diverse sources ranging from in-pile irradiation tests to operating HTGRs. In general, the agreement between predictions and measurements is encouraging; however, these design methods are not yet fully validated to the required standards. The current status and additional data needed to complete validation of these design methods are discussed.

1. Introduction

Design methods have been developed internationally over the past four decades for predicting radionuclide (RN) source terms for modular HTGRs.[1] While the evolution of these design methods is roughly comparable for prismatic and pebble-bed designs, both groups have made unique contributions to the technology; consequently, both are addressed herein.

1.1. RN control requirements for MHTGRs

Stringent, top-level RN control requirements are anticipated for future modular HTGRs as discussed in the Mechanistic Source Term White Paper (MST WP) (Idaho National Laboratory, 2010) prepared by the Next Generation Nuclear Plant (NGNP) Project. For example, for the NGNP Project, the most constraining top-level requirement is to meet the EPA Protective Action Guides at a ~400-m Exclusion Area Boundary without evacuation or sheltering. Limits on RN release from the plant and from the core that are consistent with these top-level RN control requirements are needed to derive allowable in-service fuel failure and as-manufactured fuel quality requirements to assure that these top-level requirements are met during normal plant operation and a broad spectrum of postulated accidents. To demonstrate to the regulatory authorities that these requirements will be met at a high confidence level, reliable design methods for predicting fuel performance and RN transport from the fuel to the site boundary need to be developed and validated.

1.2. MHTGR RN functional containment system

A functional containment system has been designed to limit RN release from the core to the environment to insignificant levels during normal operation and accidents. As shown schematically in Fig. 1 (Hanson, 2004), the five release barriers are: (1) the fuel kernel, (2) the particle coatings, particularly the SiC coating, (3) the fuel-compact matrix/fuel-element graphite (just the matrix for fuel spheres), (4) the primary coolant pressure boundary, and (5) the reactor building. The various RN transport phenomena in the core, primary circuit and reactor building that are modeled are also shown. The effectiveness of these barriers for containing radionuclides depends upon a number of factors, including the chemistry and half-lives of the radionuclides and the service conditions. The effectiveness of these barriers is also event-specific.

As described in the Preliminary Information Safety Document (PSID) (Stone & Webster Corporation, 1992) for the 350 MW(t) prismatic steam-cycle MHTGR, three classes of accidents are typically bounding: (1) large water ingress plus pressure relief, (2) depressurized core conduction cooldown, and (3) rapid helium depressurization. (The latter two classes are both depressurized core heatup events; the difference is that the former is initiated by a small primary coolant leak and the latter by a larger coolant leak.) In general, classes (2) and (3) are characterized by a prompt release of radionuclides from the primary coolant circuit (the circulating activity plus fractional reentrainment of the plateout activity) and a delayed release (incremental release from the core as the fuel temperatures rise). Typically, the predicted off-site doses are dominated by the delayed release, principally from I-131.

The above accident classes are listed in order of descending off-site

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[1] "Source term" used in the broadest sense to mean the radioactive source for all design functions rather than just RN release to the environment during accidents.

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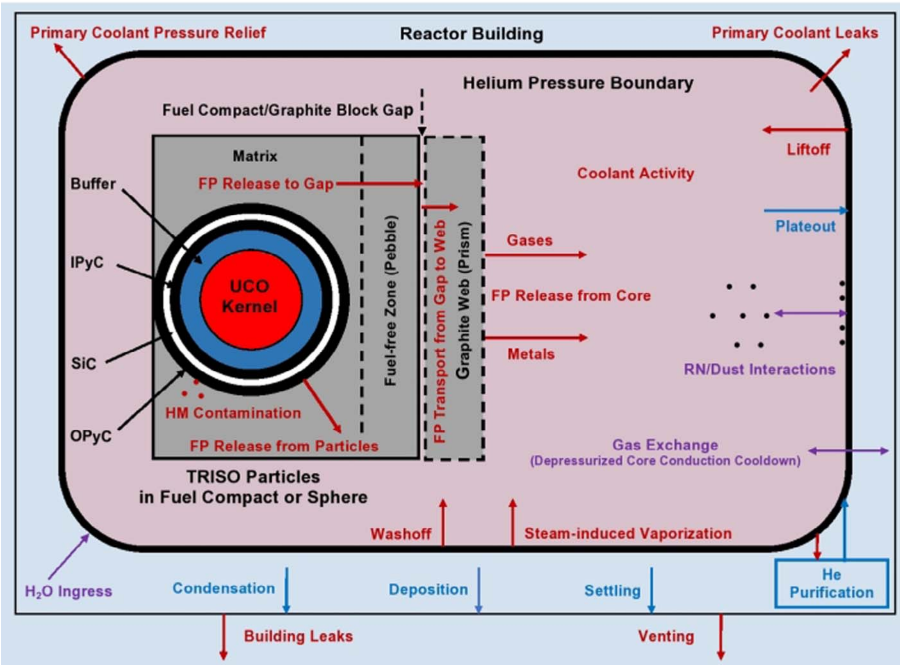

Fig. 1. MHTGR RN release barriers.

dose consequences. Hence, the dominant risk event for a steam-cycle modular HTGR is large water ingress plus pressure relief. Both depressurized core heatup classes include the introduction of variable amounts of air into the reactor vessel. Thus, all three classes of bounding events involve the fuel being exposed to oxidizing atmospheres at elevated temperatures.

The controlling RN transport phenomena for each of these accident classes are described in the NGNP MST WP, and validated design methods are needed to model these phenomena (Section 2). Hence, these events provide the technical basis for assigning priorities for code validation efforts (Section 3) and for conducting additional technology development as needed to complete the validation database (Section 4).

1.3. Predictive accuracy goals

Standard design practice in the US HTGR programs has been to define a two-tier set of RN design criteria (or allowable core releases for normal operation and anticipated operational occurrences) that are referred to as "Maximum Expected" and "Design" criteria (Stone & Webster Corporation, 1992). The "Design" criteria are derived from externally imposed requirements, such as site-boundary dose limits, occupational exposure limits, etc.; in principle, any of these RN control requirements could be the most constraining for a given reactor design.

Once the "Design" criteria have been derived from the RN control requirements, the corresponding "Maximum Expected," criteria are derived by dividing the "Design" criteria by a safety factor, or design margin, to account for uncertainties in the design methods. The fuel and core are to be designed such that there is at least a 50% probability that the fission product (FP) release from the core will be less than the "Maximum Expected" criteria and at least a 95% probability that the FP release will be less than the "Design" criteria.[2] The approach is illustrated in Fig. 2.

In the example the Preliminary Design predictions (solid lines) exceed the criteria (double lines) at the both 50% and 95% confidence levels: i.e., the nominal (50% confident) prediction exceeds the "Maximum Expected" criterion, and the upper bound (95% confident) prediction exceeds the "Design" criterion because of large uncertainties in the predictive methods. By Final Design (dashed lines), the nominal prediction meets the "Maximum Expected" criterion because of design optimization, and the upper bound prediction meets the "Design" criterion because of technology development resulting in more accurate predictive methods.

These RN design criteria also provide a logical basis for deriving predictive accuracy goals for the design methods used to predict RN source terms. The objective is to produce an optimal design which meets all requirements with sufficient, but not excessive, design margin (i.e., a tradeoff between the cost of plant design margin and the cost of technology development). In response, the accuracy goals for predicting fuel performance and FP release from the core shown in Fig. 2 were defined. An analogous approach was used to define the accuracy goals for the design methods used to predict RN transport in the primary coolant circuit and in the reactor building.

It is noteworthy that these predictive accuracy goals are not extremely precise (e.g., within an order of magnitude). Given the complexity of the phenomena that govern TRISO fuel performance and RN transport, it is neither practical nor necessary to require highly accurate predictive methods like those that are in fact required for other design functions (e.g., predicting k_{eff} or other such core physics parameters). To develop and validate such highly accurate methods for predicting source terms would be prohibitively expensive and excessively prolonged. Instead, the approach is to accept a certain level of residual uncertainty in the design methods and to include sufficient margin in the plant design to accommodate that uncertainty with high confidence. The accuracy goals shown in Fig. 2 are those adopted by the NGNP Project for a steam-cycle modular HTGR. As shown in Table 1, the PBMR Project adopted similar accuracy goals for their steam-cycle design as well (Idaho National Laboratory, 2010).

1.4. Validation of design methods

The computational methods used to support the design and licensing of nuclear power plants must be verified and validated (V&V)[3]

[2] The terms "radionuclide" and "fission product" are used interchangeably here although the former is more generic.

[3] Concisely, "verification" means confirmation of the mathematics of a computer code, and "validation" means confirmation of the physics of the code.