Generic BWR PRA Model

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The U.S. Nuclear Regulatory Commission, in conjunction with the Polish nuclear regulatory agency, has developed a probabilistic risk assessment (PRA) model for a General Electric (GE) boiling water reactor four (BWR 4). This model does not represent any specific plant, but rather is intended to be representative of a typical BWR. This report provides a discussion of the model, as of September 1, 2018. The model was developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) code.

This model is in-progress and is intended to be a training and development opportunity for domestic and international stakeholders. As other initiators, fault trees, and system failures are added, the model will be updated[1].

This report discusses the current scope of the model, internal initiating events categories, common-cause failure, and human failure events. More information on these events, as well as additional documentation of the details of the model, can be found in the model documentation.

Plant Definition and Scope

Name	Plant-Y
Type	GE BWR 4
# of Units	1 unit on the site
Containment Type	Mark II

The modeled plant design is based on the GE's Boiling Water Reactor technology BWR 4. The design used in the model is described in "General Electric Systems Technology Manual" (Reference: GEBWR-2011). The modeled plant is a one-unit plant. The design is the BWR with forced circulation of coolant, equipped with the Mark II containment.

Initiating Events

This section discusses the three initiating event categories used in the internal events model.

IE-LLOCA Initiating event large loss of coolant accident (LOCA). The large LOCA initiating event is defined as a steam or liquid break that will rapidly depressurize the reactor vessel. High-pressure injection systems will not have adequate flow rates or steam pressure to restore level and maintain core cooling.

IE-LOOP (Total) Loss of Offsite Power initiating event is a special case of a transient in which AC power from offsite power is lost to the plant emergency buses. In special cases where AC power is available to some non-safety buses, no credit is taken for

[1] This model currently has a very limited set of internal events categories and does not have any other hazard categories represented in the model. The scope of the model is substantially less detailed than the NRC's standardized plant assesstment risk (SPAR) models.

those non-safety buses. The LOOP event will cause a reactor trip. Given a LOOP event, onsite emergency diesel generators are required to start and supply emergency power to the division buses for the safety equipment. If the emergency diesel generators fail, then a station blackout (SBO) event will occur. The SBO is modeled in a separate event tree. Failure of reactor protection system to shutdown the reactor will cause this sequence to transfer to the anticipated transient without scram (ATWS) event tree for further evaluation.

IE-TRANS This initiating event category includes common transients, such as a turbine trip and spurious reactor trip, where the secondary heat removal through the condenser and main feedwater system is available at the time of the reactor trip. Failure of reactor protection system to shutdown the reactor will cause this sequence to transfer to the ATWS event tree for further evaluation.

Common Cause Modeling

Generally, common cause failures are modeled for active components, except for plugging of heat exchangers. Common cause failure is only modeled for like components within a system, and not across system boundaries. Common cause modeling is considered for the following types of components:

- motor-operated valves,
- air-operated valves,
- explosive valves,
- safety relief valves, power-operated relief valves (PORV),
- check valves,
- pumps, heat exchangers and
- diesel generators.

Common cause failure probability is calculated using the Alpha Factor Method.

Human Failure Events

The human actions included in the model consist of both pre-accident failures to restore systems following test or maintenance, and post-accident failures to align systems, to control or operate systems, and to recover system hardware failures. In the current version of the model a detailed human reliability analysis (HRA) has not been developed. Human error probabilities (HEPs) used in the current version of the model are not justified by any HRA method. All HEPs have a generic value of 10E-3 assigned. In some cases, lower or higher values have been used to address predicted significant differences in the time available to perform action.

Quantification of Total Plant CDF from All Hazard Categories

The total plant CDF from all scenarios is calculated by selecting all scenarios in the SAPHIRE The overall Core Damage Frequency (CDF) for PLANT-Y model limited to three ETs, representing accident models for three IEs (LLOCA, LOOP-WR, TRANS) is following:

CDF (Point Estimate):

- Non-minimal cutsets quantification: **1.87E-05**

- Minimal cutsets quantification: **1.81E-05**

As noted previously, this version of the model does not contain what would be considered as a complete set of initiating events for any of the hazard categories modeled; however, as the model is updated and refined, these limitations could be resolved.

References

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