Generic PWR PRA Model

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The U.S. Nuclear Regulatory Commission, in conjunction with the Nuclear Regulation Authority, Japan, has developed a probabilistic risk assessment (PRA) model for a 4-loop pressurized water reactor (PWR). This model does not represent any specific plant, but rather is intended to be representative of a simplified PWR. This report provides a discussion of Version 3.1 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.

A summary of the core damage frequency (CDF) for Version 3.1 has been calculated and output tables generated by using the following:

Model Version: PLANT-X Version 3.1 SAPHIRE Version: 8.1.8 Frequency Cutoff: 1E-13 Number of Event Trees (ETs): 21 Run Time = 1 minute
Output Summary for Version 3.1 With Non-minimal CDF/number of cutsets = 6.803E-05 / 146046 With Minimal CDF / number of cutsets = 6.660E-05 / 129375

This report discusses the scope of the model, internal initating events categories, common-cause failure, and human failure events. The model does include, on a limited basis, other hazard categories beyond internal 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-X
Type	4-loop PWR
# of Units	1 unit on the site
RCP Seals	WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs
SBO DG	1 SBO DG
ELAP/FLEX	Procedures exist for Total Loss of AC Power

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, including the internal events[1]. The scenarios (initiating events) for other hazard categories use as much as possible the already

[1] This model currently has a limited set of internal events categories and a few scenarios representing some of the other hazard categories applicable to the site. The scope of the model is substantially less detailed than the NRC’s standardized plant assesstment risk (SPAR) models.

modeled internal events. This is a plant CDF model; LERF is not calculated, although SAPHIRE can accommodate a simple modeling extention to LERF. Events during shutdown operations are not included in this model.

Initiating Events

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

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 anticicpated transient without scram (ATWS) event tree for further evaluation.

IE-LLOCA: The large loss of coolant accident (LOCA) initiating event is defined as a steam or liquid break that is large enough to rapidly depressurize the reactor coolant system (RCS) pressure to a point below the low pressure injection and accumulator shutoff pressure. This break size is generally defined as being greater than 5 inches.

IE-MLOCA: The medium LOCA initiating event is defined as a steam or liquid break that is large enough to remove decay heat without using the steam generators but small enough that RCS pressure is above the accumulator and low pressure injection system shutoff pressure.

IE-SLOCA: The small LOCA initiating event is defined as a steam or liquid break in the RCS other than a steam generator tube rupture which exceeds normal charging flow. In this break size range, normally defined as between 3/8 inches and 2 inches, normal charging cannot maintain pressurizer level. A small LOCA will depressurize the RCS and cause a reactor trip. A safety injection signal will also be generated to start the high-pressure injection (HPI) pumps. Secondary cooling is required to remove decay heat and cause the RCS to reach an equilibrium pressure which corresponds to the injection flow of the HPI pumps. Since primary pressure is above the HPI shutoff head, secondary cooling is required. If secondary cooling fails, then feed and bleed cooling is required to remove decay heat.

IE-LOOP: (Total) Loss of offsite power (LOOP) 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 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. 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 ATWS event tree for further evaluation.

IE-LMFW: Transients with loss of main feedwater (MFW): this event is the same as TRANS initiating event except that MFW is lost and is assumed not to be recoverable.

Common Cause Modeling

As a general rule, 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,
- safety relief valves,
- power-operated relief valves (PORV),
- check valves,
- pumps,
- heat exchangers,
- containment sump plugging and
- diesel generators.

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

Human Failure Events

The operator 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. Pre-accident failures to restore systems following test or maintenance are quantified using generic Accident Sequence Evaluation Program Human Reliability Analysis Procedure, (ASEP) data, data from NUREG-1150, and engineering judgment.

The Human Error Probabilities (HEPs) for the modelded actions were obtained from a number of sources as noted. Some of the HEPs were obtained from previous NRC sponsored studies as noted. The remaining HEPs were calculated using the SPAR Model Human Reliability Analysis (HRA) method described in NRC-2005-02. The SPAR-H method builds on the Technique for Human Error Rate Prediction (THERP) method outlined in NUREG/CR-1983. The HEPs calculated using the SPAR HRA method are documented using Human Error Worksheets that are provided in NRC-2005-02. The Human Error Worksheets are used to evaluate operator actions with regard to various characteristics including stress and threat level, time available to respond, training, procedure quality, dependence, and experience.

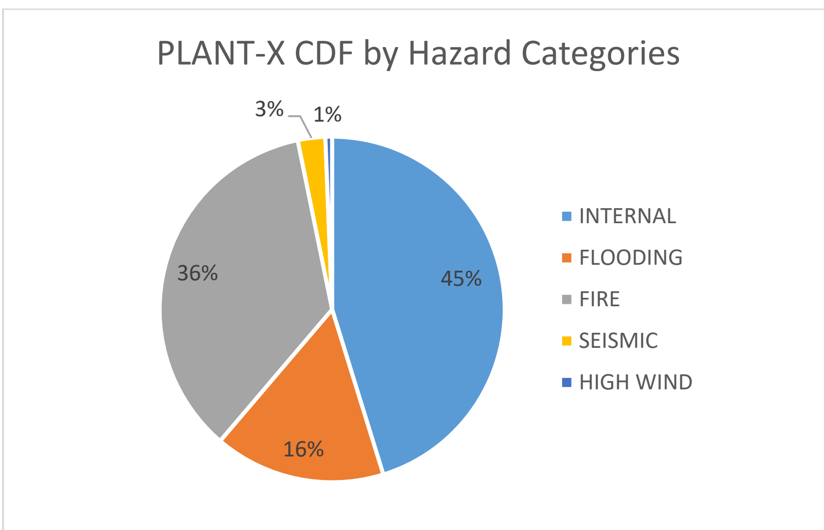
Since the HEP of many of the operator actions in the model may be dependent on the success or failure of other operator actions, a given operator action may have different HEP values in different cut sets. The SPAR HRA method has provisions for calculating the HEP for each context using a formal dependency calculation; however, the number of combinations that must be addressed is immense. To keep the number of dependency calculations that must be made to a manageable number, the dependency calculation between human failure events was simplified to include only the operator actions involving the actuation, alignment, or control of systems that perform similar, usually redundant, functions.

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 Event Tree window, and solving for them. All scenarios are marked as the RANDOM model. Other model types are not used.

The below table and figure show the results for all hazards CDF.

	Hazard Category	**# of IEs**	**IE Frequency**	**CCDP**	**CDF**	**# of Cut sets**	**# CDF**
1	INTERNAL EVENTS	15	8.23E-01		3.01E-05	1.10E+05	45.2%
2	INTERNAL FLOODING EVENTS	1	3.00E-04	3.56E-03	1.07E-05	3.56E+02	16.0%
3	INTERNAL FIRE EVENTS	1	1.00E-04	2.37E-01	2.37E-05	1.06E+03	35.6%
4	SEISMIC EVENTS	3	3.24E-06	5.30E-01	1.71E-06	3.62E+03	2.6%
5	HIGH WIND and TORNADO EVENTS	1	1.00E-03	4.14E-07	4.14E-07	1.46E+03	0.6%
	Total =	21	8.24E-01		6.66E-05	1.29E+05	100%



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 can be resolved.

References

IAEA-1993 Defining initiating events for purposes of probabilistic safety assessment - IAEA Publications
www-pub.iaea.org/MTCD/publications/PDF/te_719_web.pdf

INL-2011 SPAR-H Step-by-Step Guidance, Galyean, W.J., Whaley, A.M., Kelly, D.L., Boring, R.L. Idaho National Laboratory, May 2011.
https://www.nrc.gov/docs/ML1120/ML112060305.pdf

NRC-1981 NUREG-0492. Fault Tree Handbook
https://www.nrc.gov/docs/ML1007/ML100780465.pdf

NRC-1983 Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (THERP), NUREG/CR-1278, August.
https://www.nrc.gov/docs/ML0712/ML071210299.pdf

NRC-1998 NUREG/CR-6544, A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences
https://www.nrc.gov/docs/ML0716/ML071650470.pdf

NRC-2002-02 Part 2 of 4 - Westinghouse Technology Manual, Course Outline ... - NRC
https://www.nrc.gov/docs/ML0230/ML023040145.pdf

NRC-2002-03 Part 3 of 4 - Westinghouse Technology Manual, Course Outline ... - NRC
https://www.nrc.gov/docs/ML0230/ML023040213.pdf

NRC-2003 SAFETY EVALUATION OF TOPICAL REPORT WCAP-15603, REVISION 1, "WOG 2000 REACTOR COOLANT PUMP SEAL LEAKAGE MODEL FOR WESTINGHOUSE PWRS"
https://www.nrc.gov/docs/ML0314/ML031400376.pdf

NRC-2005-01 NUREG-1792, Good Practices for Implementing Human Reliability Analysis, 2005
https://www.nrc.gov/docs/ML0511/ML051160213.pdf

NRC-2005-02 SPAR-H Human Reliability Analysis Method, NUREG/CR-6883, August 2005.
https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6883/

NRC-2007-01 NUREG/CR-6928. INL/EXT-06-11119. Industry-Average Performance for Components and Initiating. Events at U.S. Commercial. Nuclear Power Plants.https://www.nrc.gov/docs/ML0706/ML070650650.pdf

NRC-2007-02 Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007 (NUREG/CR-7037, INL/EXT-10-17932)
https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7037/

NRC-2008 NUREG-1829 Vol. 1 Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process Main Report
https://www.nrc.gov/docs/ML0822/ML082250436.pdf

NRC-2012 Westinghouse Technology Systems Manual Section 7.2 Condensate and Feedwater System https://www.nrc.gov/docs/ML1122/ML11223A246.pdf

NRC-2013-01 Risk Assessment of Operational Events Handbook Volume 1 – Internal Eventshttps://www.nrc.gov/docs/ML1303/ML13030A049.pdf

NRC-2013-02 Risk Assessment of Operational Events Handbook Volume 2 – External Eventshttps://www.nrc.gov/docs/ML0803/ML080300179.pdf

NRC-2013-03 Risk Assessment of Operational Events Handbook Volume 3 – SPAR Model Reviewshttps://www.nrc.gov/docs/ML1028/ML102850267.pdf

NRC-2015-01 INL/EXT-14-31428. Initiating Event Rates at U.S. Nuclear Power Plants 1988–2013 https://nrcoe.inel.gov/resultsdb/publicdocs/InitEvent/initiating-event-frequencies-and-trends-2013.pdf

NRC-2015-02 CCF Parameter Estimations 2015. https://nrcoe.inel.gov/resultsdb/publicdocs/CCF/ccfparamest2015.pdf

NRC-2015-03 Parameter Estimates Spreadsheet 2015. http://nrcoe.inl.gov/resultsdb/publicdocs/AvgPerf/ParameterEstimates2015.xlsx

NRC-2016-01 SPAR Initiating Event Data and Results 2015 Parameter Estimation Update, 2016 http://nrcoe.inel.gov/resultsdb/publicdocs/AvgPerf/InitiatingEvents2015.pdf

NRC-2016-02 Analysis of Loss-of-Offsite-Power Events 1987–2015 INL/EXT-16-39575, 2016. https://nrcoe.inel.gov/resultsdb/publicdocs/LOSP/loop-summary-update-2015.pdf