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Markov Model of Severe Accident Progression and Management

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Abstract: The earthquake and tsunami that hit the nuclear power plants at the Fukushima Daiichi site in March 2011 led to extensive fuel damage, including possible fuel melting, slumping, and relocation at the affected reactors. A so-called feed-and-bleed mode of reactor cooling was initially established to remove decay heat. The plan was to eventually switch over to a recirculation cooling system. Failure of feed and bleed was a possibility during the interim period. Furthermore, even if recirculation was established, there was a possibility of its subsequent failure. Decay heat has to be sufficiently removed to prevent further core degradation. To understand the possible evolution of the accident conditions and to have a tool for potential future hypothetical evaluations of accidents at other nuclear facilities, a Markov model of the state of the reactors was constructed in the immediate aftermath of the accident and was executed under different assumptions of potential future challenges. This work was performed at the request of the U.S. Department of Energy to explore "what-if" scenarios in the immediate aftermath of the accident. The work began in mid-March and continued until mid-May 2011. The analysis had the following goals:

- To provide an overall framework for describing possible future states of the damaged reactors,
- To permit an impact analysis of "what-if" scenarios that could lead to more severe outcomes,
- To determine approximate probabilities of alternative end-states under various assumptions about failure and repair times of cooling systems, and
- To infer the reliability requirements of closed loop cooling systems needed to achieve stable core end-states and
- To establish the importance for the results of the various cooling system and physical phenomenological parameters via sensitivity calculations.

Keywords: Accident Management, Severe Accidents, Markov Model

1. INTRODUCTION

A severe magnitude 9.0 earthquake followed by a tsunami on March 11, 2011, that caused loss of offsite power and disabled the emergency diesel generators, led to a prolonged station blackout at the Fukushima Daiichi site. While reactor trip was successful for all operating reactors, the inability to remove decay heat over an extended period led to boil-off of the water inventory and fuel uncovery in Units 1, 2, and 3. A significant amount of metal-water (zirconium oxidation) reaction occurred as evidenced by the quantities of hydrogen generated that led to a hydrogen explosion in the auxiliary buildings of the Units 1 and 3. A hydrogen explosion also occurred at the de-fuelled Unit 4 and it has been speculated that this was due to hydrogen migrating from Unit 3. Units 5 and 6 were not fuelled at the time of the earthquake and were also at a higher elevation and thus did not suffer the consequences of the other units. Although it was assumed that extensive fuel damage, including fuel melting, slumping, and relocation was likely to have occurred in the core of the affected reactors, the status of the fuel, vessel, and drywell was uncertain. It is not clear, for example, if the core was retained in the vessel or if vessel breach had occurred in any of the units, allowing corium to spread on the floor and possibly leading to core-concrete interaction and further generation of non-condensable gases pressurizing the containment.

Remedial measures for injection to cover and cool the core were started initially with diesel-driven fire pumps using seawater. Since the normal mode of removing heat with the reactor shutdown, i.e., through the residual heat removal system, was disabled by the event, injection through the feedwater lines had initially taken place and the core was cooled by feed-and-bleed (F&B) procedure, where heat was removed, as far as is known, by first releasing steam to the containment and then venting through the wetwell. It is likely that such venting received some benefit of scrubbing as the release went through the suppression pool, although the amount of scrubbing is very uncertain. It is also possible that some venting bypassed the suppression pool altogether or went directly out of (unknown) damaged locations in the drywell. What is known is that highly contaminated water leaked into sumps in the turbine building. This suggests leaks in the wetwell(s) or, perhaps, in locations in the drywell, as well as leaks in the reactor vessel, that allowed the injected water to leak into the turbine building.

2. OBJECTIVES

To understand the possible evolution of the accident conditions at Fukushima Daiichi, a Markov model [1] of the likely state of one of the reactors was constructed and was executed under different assumptions of future challenges. This work was performed at the request of the U.S. Department of Energy to explore "what-if" scenarios in the immediate aftermath of the accidents. The work began in mid-March and continued until mid-May 2011.The analysis had several goals:

1. To provide an overall framework for describing possible future states of the damaged reactors,
2. To permit an impact analysis of "what-if" scenarios that could lead to more severe outcomes,
3. To determine approximate probabilities of alternative end-states under various assumptions about failure and repair times of cooling systems,
4. To infer the reliability requirements of closed loop cooling systems needed to achieve stable core end-states and
5. To establish the importance for the results of the various cooling system and physical phenomenological parameters via sensitivity calculations.

The Markov approach was selected for several reasons: It is a probabilistic model that provides flexibility in scenario construction and incorporates time dependence of different model states. It also readily allows for sensitivity and uncertainty analyses of different failure and repair rates under various assumptions regarding system performance and reliability. While the analysis was motivated by a need to gain insight on the course of events for the damaged units at Fukushima Daiichi, the work reported here provides a more general analytical basis for studying and evaluating the accident evolution over extended periods of weeks and months.

While some early consideration was given to the condition of the spent fuel pools, the main emphasis of the work was on the damaged reactors.

3. ASSUMPTIONS USED IN CONSTRUCTING THE MARKOV MODEL

The first step in model construction was to provide a representation of the states of the core and containment and the transitions between these states that could take place due to potential failures or repairs of the cooling systems that remove heat from the damaged core and containment. A generic model with damaged fuel inside the reactor vessel as the initial state was taken to represent, approximately, the initial state of any of the damaged units.

A limited amount of cooling with venting of the containment is assumed. For the fuel still in-vessel this is called the feed & bleed (F&B) mode. If the vessel is breached and the fuel has penetrated ex-vessel, it is called the flood & vent (F&V) mode.

It was assumed, at the time of this work, that plant managers were attempting to construct a closed-loop recirculation system to remove heat. When this would be established, cooling would be provided by recirculation. This is definitely the preferred method for cooling the core as it precludes venting and consequent release of fission products into the secondary buildings and the outside environment, and also prevents leakage of highly contaminated water into sumps in the turbine building and eventually outside the plant.

Different fission product release states would be identified with the various nodes in the model describing the states and location of the core. These are expected to be associated with different radiological signatures (noble gases, volatiles, non-volatiles, etc.). The following fission product release states were distinguished:

- In-vessel release with controlled containment venting (scrubbed release)
- In-vessel release with containment venting and no scrubbing
- In-vessel release with a breached containment
- Ex-vessel release with breached wetwell
- Ex-vessel release with breached drywell
- Uncontrolled release with corium spread outside containment

The analysis could be suitably adapted as a guide to decision-making in post-accident situations where a variety of different strategies may need to be assessed as prolonged offsite release may occur in response to both external and internal challenges.

If cooling to the reactor vessel is lost for any reason, then, given the damaged core in-vessel and the existing decay heat level (i.e. about 2 months after reactor scram), the core will melt and relocate to the bottom of the vessel, start to attack the bottom head and possibly penetrate the vessel to slump to the floor of the drywell. This will initiate the core-concrete interaction as the molten corium spreads on the drywell floor, leading possibly to drywell failure from a number of different mechanisms if no further injection occurs. The approximate timing of key events is based on analysis of a similar BWR reactor for which severe accident evaluations were performed several years ago [2] as part of the severe accident research program in the USA. In the current paper, no distinction is made among the reactor units at the Fukushima Daiichi site. Rather, the timing and meltdown characteristics are those for the typical BWR. In the first approximation and for the purposes of this paper, this is a reasonable assumption. The typical accident progression is shown in Table 1 below (accounting for reduced decay heat 2 months after scram):

Table 1: Timing of Key Events in Accident Progression

Event	Time after loss of cooling (hours)
Loss of injection	0.0
Core uncovery	20
Start melt	33
Core slump/collapse	39
Bottom head failure	44
Start concrete attack	44
Containment failure	65

These timings (and others presented below) were estimated based upon previous experience in modelling and calculating similar core meltdown scenarios, and were utilized to estimate the transition time between different states of the core, degraded core with F&B cooling and molten fuel on the bottom head, and the core location (in-vessel or ex-vessel). As time goes on, these accident progression times become more delayed and stretched out due to lower decay heat level. This would be recognized in accident modeling for failure of cooling events occurring at later times (e.g. on the order of months into the future).

Each core state represents a change in its physical condition in the accident progression. The in-vessel states cover core heatup through vessel failure. The ex-vessel states represent the fuel debris on the drywell floor and associated phenomena like debris spread, generation of non-condensable gases like carbon dioxide, carbon monoxide, and hydrogen through the core concrete interaction, as well as possible mechanisms of containment (drywell) failure such as overpressurization or liner melt through. It was assumed at the start of the analysis that the core was already degraded, and it was very likely that in some of the units the vessel did not have a full inventory of water as there were possibly leaks in the vessel. In that event, it was quite possible that the time to start core melt and cause the core to slump on the bottom head could be considerably shorter than shown in the above table if all injection was lost. The uncertain current status of the core and the water level in the timing of the key events in the accident progression was addressed through sensitivity calculations.

Table 2 shows sensitivity ranges for the times of failure and recovery of the cooling systems, assumed to set up the baseline model. All times are given in days. These include a reference case and worst and best cases, which were used for sensitivity studies, for the various parameters:

Table 2: Key Transition Rates for in-Vessel Markov Model Development

	Reference case	Worst case	Best Case
Recir. failure	500	100	900
Recirc. recovery	45	60	30
F&B failure	60	30	150
F&B recovery	1	5	0.25
Fuel to bottom head	1.63	1	3
Fuel outside vessel	0.19	0.1	1

It should be noted that these reference values were based on very approximate estimates that were used to develop the model.

4. MARKOV MODEL DEVELOPMENT

The basis for the in-vessel Markov model is shown in Figure 1. The transition rates [1] between events or states shown in this figure are derived from Table 2. As stated earlier, the starting point was the assumption of a degraded core that was cooled by an F&B cooling with venting cooling mode. The baseline model was set up under the assumption of random failures of the cooling system, with mean time to failure and recovery rates as shown above.

The loss of F&B cooling triggers another node that can then progress on to other nodes representing more severe events in the accident progression, or not, depending on the competition between restoration of F&B and the timing of core degradation, melt and slump events. After recirculation cooling is established, F&B cooling with venting would become a kind of backup cooling system to prevent core melt and vessel breach scenarios. This is shown in Figure 1 in the lines connecting the various nodes that depict transitions from one state to another. The failure and repair rates, as stated earlier, are approximate, to allow the model to be run and provide intuitively useful results that give confidence in the basic setup of the model.

The in-vessel model was run to examine the probabilities of different states of the core as a function of time. The time scale over which the initial runs were made is on the order of one to two years.

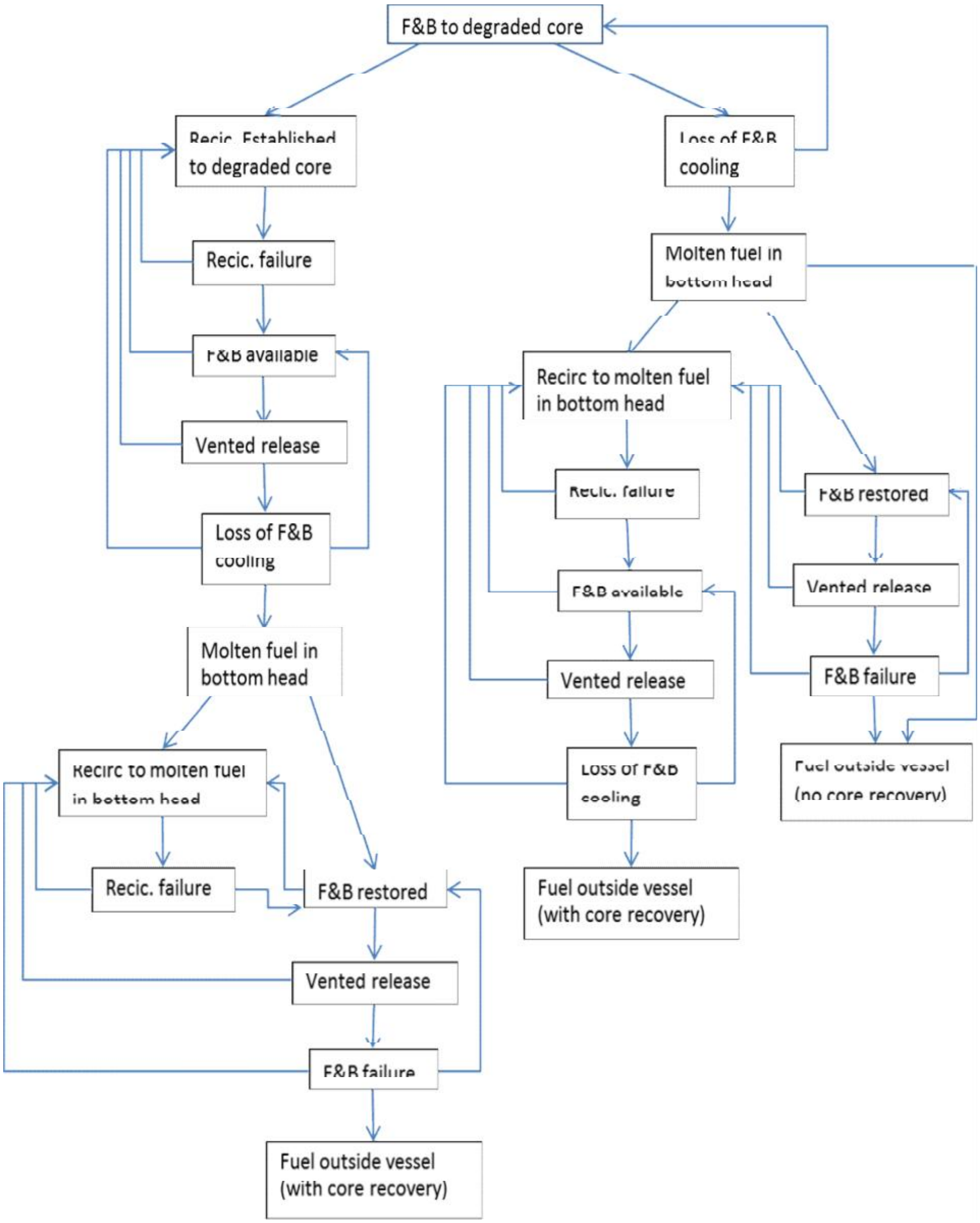

Figure 1 - Markov in-vessel model

5. RESULTS

A check of the Markov model logic was performed by calculating and comparing various cases using the parameters in Table 2 above. For one case the reference values were used for all the parameters. The other two cases were a "worst" case and a "best" case. For the worst case the low values of the ranges in Table 2 were chosen for the mean times to equipment failure and core melt progression, while the high values of the range for equipment recovery were chosen. For the best case the opposite was true: the high range values were input for equipment failure and accident progression and the low times were input for equipment recovery.

The variation in the results between the three cases showed the expected trends. For example, as it can be seen in Figure 2, at 150 days the reference case showed about a 0.21 probability of fuel relocating ex-vessel, compared to a 0.85 probability for the worst case and essentially a zero probability for the best case. Similarly, the probability of being in a stable recirculation cooling mode at 150 days inferred from Figure 3 was about 0.75 for the reference case, 0.17 for the worst case and essentially 1.0 for the best case. From Figure 4, the probability of being in a feed and bleed mode by 150 days was only a few percent for the reference and worst case, and negligible for the best case. These results appear reasonable given the parameters chosen.

Sensitivity runs were performed to establish the importance for the results of the various parameters. These parametric studies shed light on what parameters play a vital role in minimizing vessel failure and migration of fuel ex-vessel, and indicated that the assumed mean time to failure of the recirculation system (500 days for the reference case, 900 days for the best case, and 100 days for the worst case) was a key parameter. For instance, Figure 5 shows that at 150 days the probability of fuel relocating ex-vessel increased from 0.21 to 0.33 when the mean time to failure was decreased from 500 days to 100 days, while it did not change much (from 0.21 to 0.20) when the mean failure time was increased to 900 days. Similarly, the probability of being in a stable recirculation cooling mode at 150 days decreased to 0.64 when the mean failure time was decreased to 100 days (Figure 6).

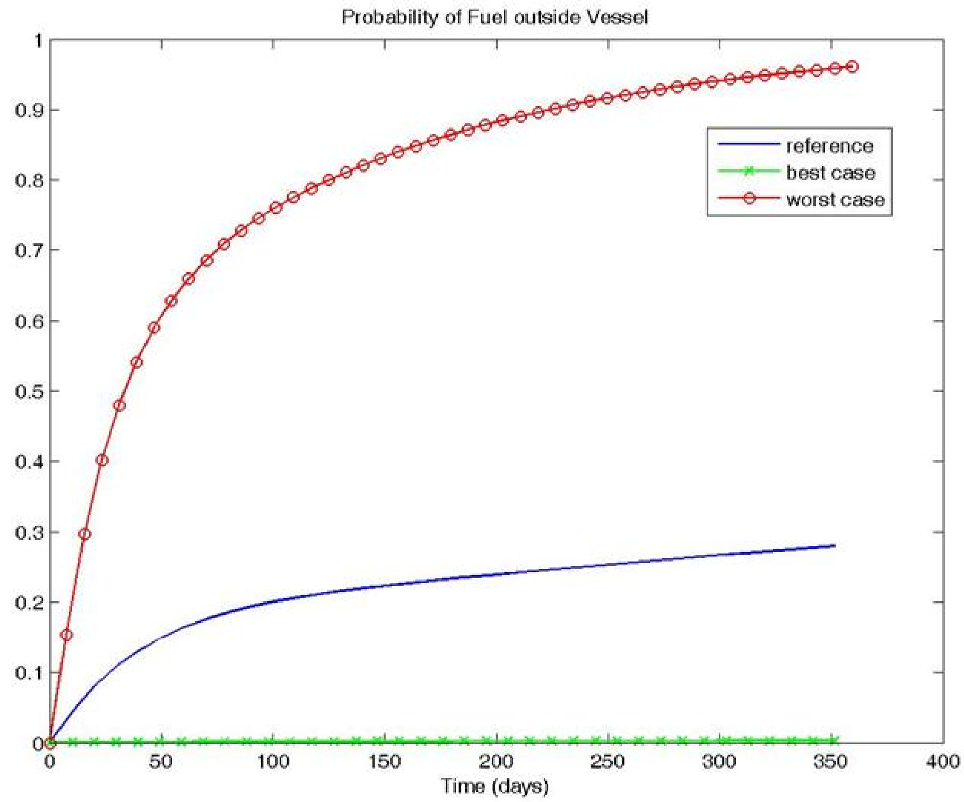

Figure 2 - Probability of fuel relocating ex-vessel

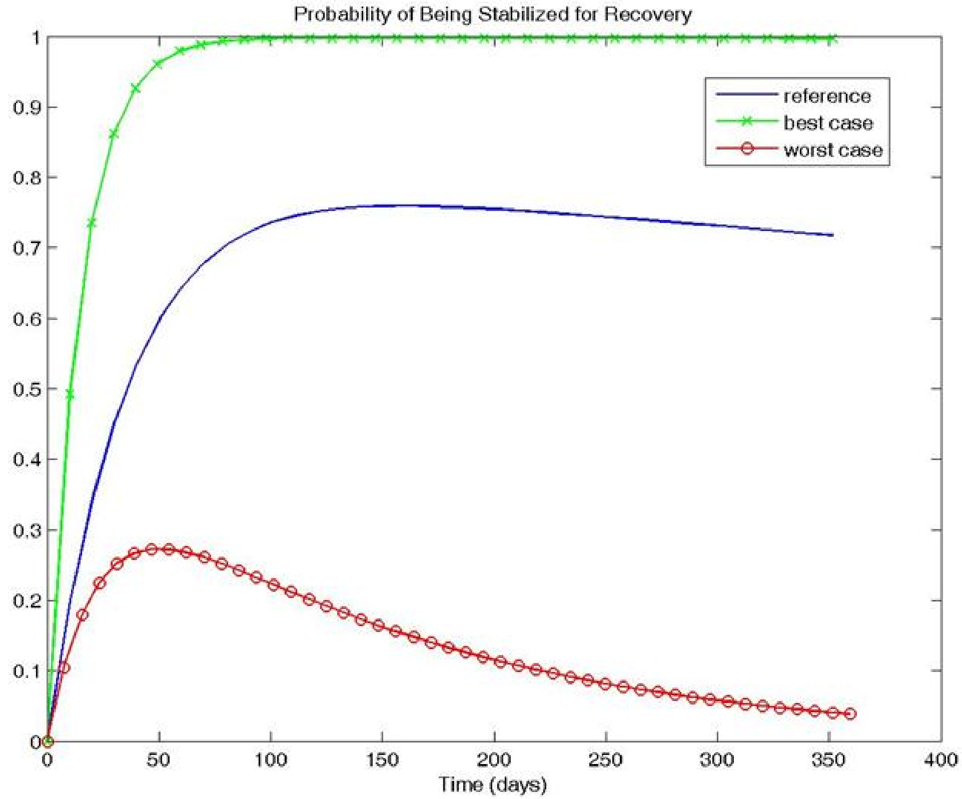

Figure 3 - Probability of being in recirculation mode

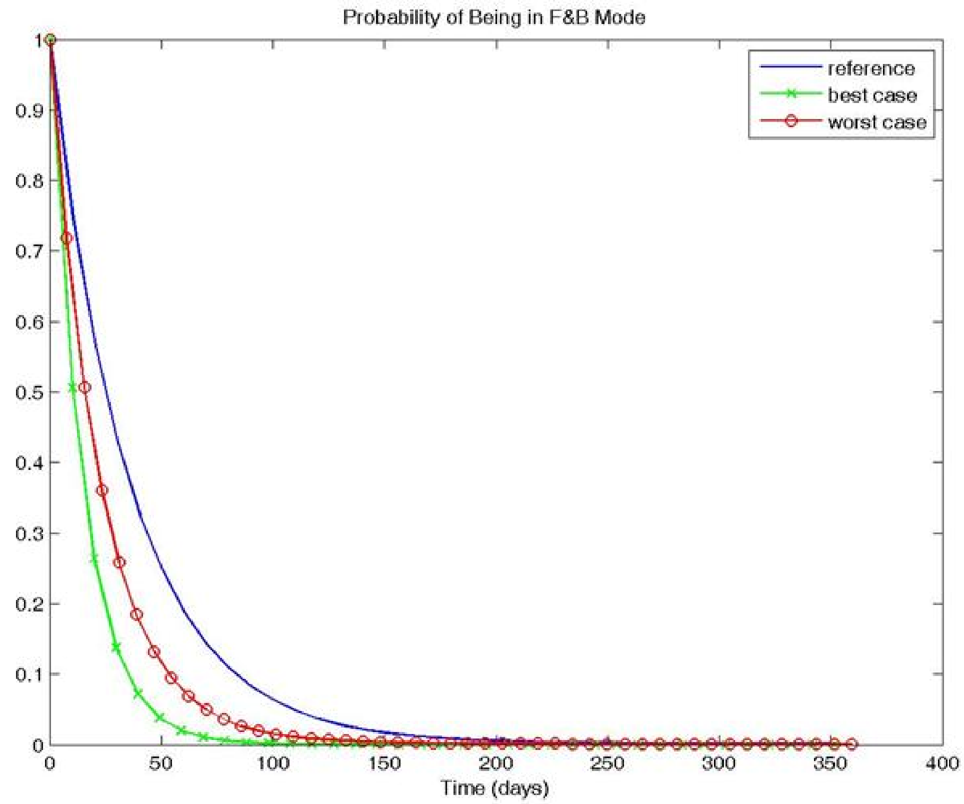

Figure 4 - Probability of being in Feed & Bleed mode

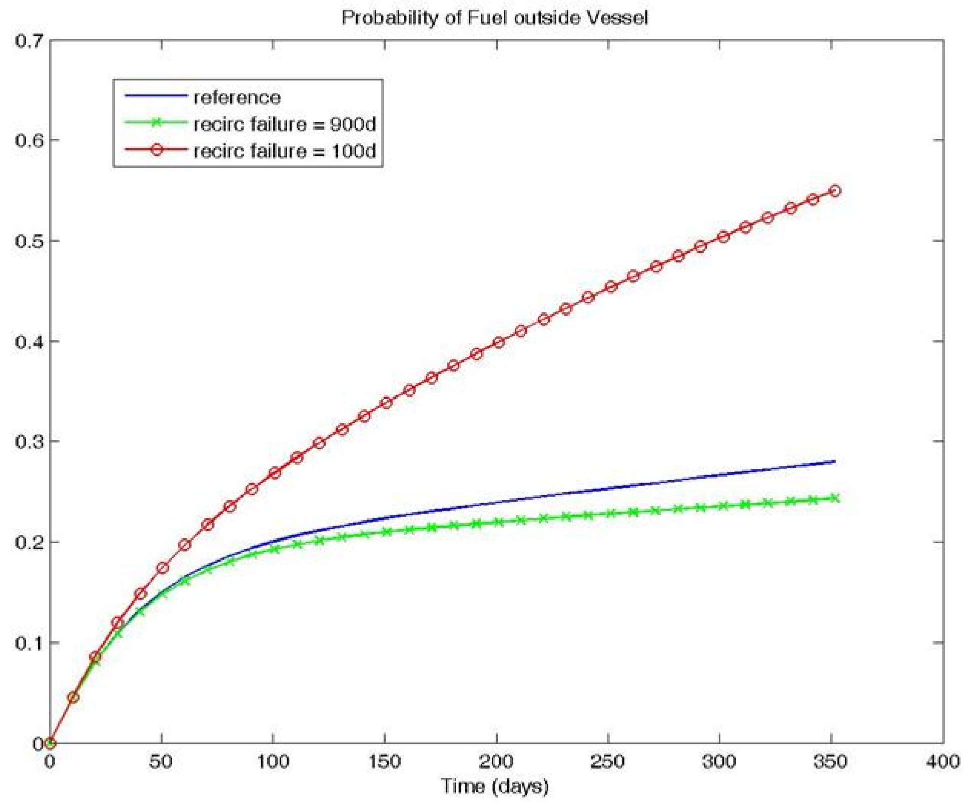

Figure 5 - Sensitivity of recirculation cooling failure time – Probability of fuel relocating ex-vessel

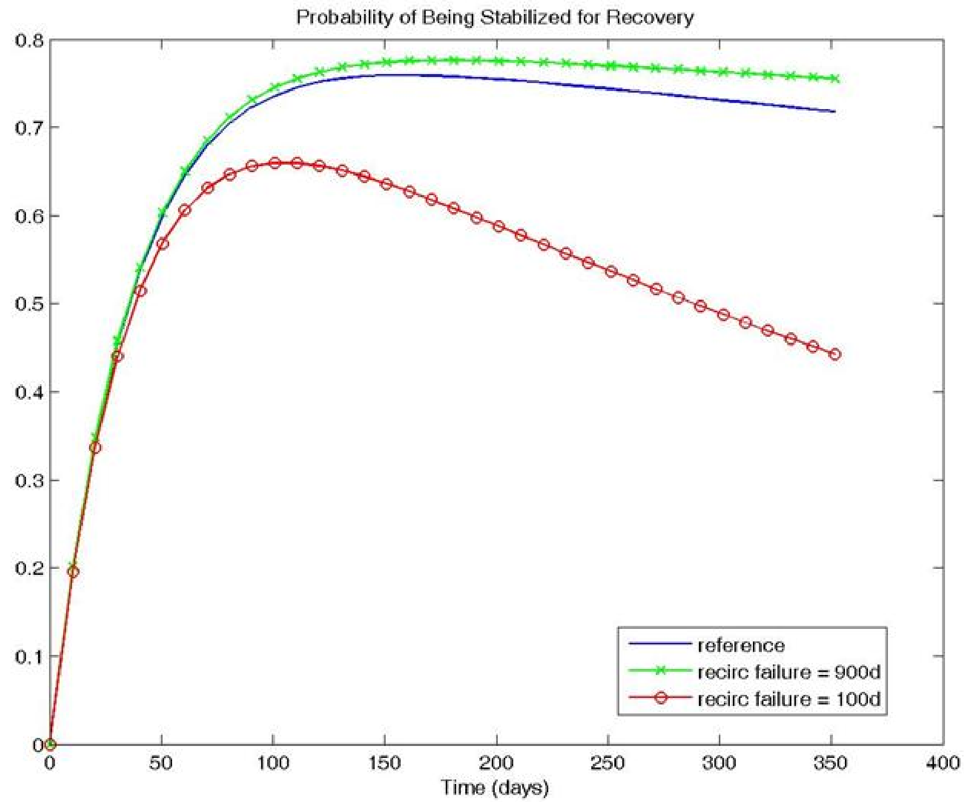

Figure 6 - Sensitivity of recirculation cooling failure time – Probability of being in recirculation mode

6. SUMMARY AND OUTLOOK

This work illustrates how a Markov model approach can be used to describe and predict outcomes of a severe accident for which much uncertainty exists. Further it enables the study of the time-dependent dynamic events associated with failure and restoration of cooling modes to be evaluated in comparison to heat-up times for damaged fuel. It can thus be used for accident management studies of other reactors if the relevant physical and engineering parameters are incorporated in the model.

In the case of Fukushima Daiichi, the what-if scenarios could be developed in such a way that the physical description of the challenge and the expected plant response are reasonably consistent. For example, if a scenario involves a severe seismic event that may involve a significant disruption of plant systems, the cooling systems could fail almost instantaneously while their recovery and repair times are expected to be significantly longer than those associated with random failures. Several months into the future, as the core cools, the time between key events in the accident progression will also increase, thus increasing the probability of achieving a stable state with the core in-vessel and the recirculation cooling system operational.

While the work report here was helpful for the authors to gain an understanding of the potential evolution of events at Fukushima Daiichi, there was essentially no interaction with the plant operators and owners during this work. Fortunately, there was no severe and unexpected downturn of events at Fukushima Daiichi after this work ended in late spring of 2011. Since then, Units 1-3 have been brought to cold shutdown with a mode of recirculation cooling established [3].

The implications of the Fukushima Daiichi accident for facilities worldwide have been evaluated by many organizations. In particular, in the U.S., in addition to the studies conduct by the U.S. Department of Energy's Office of Nuclear Energy, evaluations have been performed [4, 5, 6] by government agencies and by the private sector.

Acknowledgements

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References

[1] Gnedenkov, B., The Theory of Probability, MIR Publishers, Moscow, 1973.

[2] This study drew on meltdown characteristics that were reported in NUREG-1150 for the Peach Bottom Mark 1 Boiling Water Reactor. See "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, U.S Nuclear Regulatory Commission, December 1990.

[3] The website of the International Atomic Energy Agency provides monthly updates (and links) to the status of the Fukushima plants and their impact: www.iaea.org.

[4] For studies of nuclear facilities operated by the U.S. Department of Energy: http://www.hss.doe.gov/nuclearsafety/nsworkshop2011/.

[5] For facilities regulated by the U.S. Nuclear Regulatory Commission, see: http://pbadupws.nrc.gov/docs/ML1118/ML111861807.pdf.

[6] The timeline and narrative of the first few weeks of the accidents has been developed by the Institute of Nuclear Power Operations: Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station, INPO 11-005, November, 2011.