Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 29-30, 2014

The Maximum Coping Time Analysis of the ELAP for the OPR1000

Sung Hyun Shin[a*], Chang Joo Hah [a], Si Chae Jung [b], Chang Gyun Lee [b]
[a]*KINGS, 658-91, Haemaji-ro, Seosaeng-myeon, Ulju-gun, Ulsan, 689-882*
[b]*KEPCO E&C, 989-113, Daedeokdae-ro, Yuseong-gu, Daejeon, 305-353*
[*]*Corresponding author: Shining@kepco-enc.com*

1. Introduction

There have been many evaluations and recommendations for the extended Station Black Out (SBO) condition of the nuclear power plant. For example, the "SECY-11-0093/0137" [1], [2] is a recommendation of NRC and the "WCAP-17601-P" [3] is an evaluation of the PWROG. In response to the extraordinary events that occurred at Fukushima in Japan in March, 2011.

The extended loss of AC power (ELAP) can be defined as same with the extended (or prolonged) SBO which has a Loss of Offsite Power (LOOP) condition and loss of all Emergency Diesel Generators (EDG), Alternative Alternating Current (AAC), but Direct Current (DC) source is available.

This evaluation provides NSSS responses to an ELAP for the OPR1000 unit. And the results presented provide certain phenomena which occur during the ELAP, the maximum coping time until a core uncovery condition.

2. Methods and Results

Specifically, this study evaluates the comparative advantage of cooling and depressurizing the RCS, assuming maximum RCP seal leakage, when compared to the baseline case performed with minimal cooldown of the RCS, intended only to maintain subcooling margin of the cold leg at > 50 ^{o}F for as long as possible and maximum RCP seal leakage in the "WCAP-17601-P". For the baseline case analysis of the St. Lucie unit having a similar thermal power capacity to the OPR1000, the core remains covered for approximately 66 hours.

2.1 Scenario of the ELAP for the OPR1000

The ELAP is considered beyond Design Basis. Initiating event is a loss of off-site power (LOOP) with a concurrent loss of on-site AC power.

The following is assumed to occur at time zero due to loss of all AC power:

- Reactor and turbine trips
- RCPs coastdown
- Letdown isolates
- Charging pumps de-energize
- Proportional and backup heaters de-energize
- Pressurizer spray is unavailable
- HPSI and LPSI pumps are unavailable
- Steam dump and bypass system is unavailable
- Main feedwater pumps coastdown
- Motor driven AFW pumps are unavailable
- RCP seal leakage

2.2 Important Analysis Assumptions & Inputs

This is a best estimate analysis for beyond design basis event. Therefore, all equipments are expected to operate at nominal setpoints and capacities. Also, no single failures are modeled.

The major assumptions of the inputs are listed as follows:

① Event is initiated from full power operation.

② Decay heat is per ANS 5.1-1979 + 2 sigma, or equivalent.

③ The auxiliary feedwater supply will be provided symmetrically to all SGs.

④ If possible, the SITs will be isolated at an appropriate time to simulate the effects of venting the nitrogen cover gas from these storage tanks. This is performed such that non-condensable will not be introduced in bulk into the RCS.

⑤ Steam flow to a TDAFW pump was not modeled since this flow provides an additional steaming path that may need to be considered for long term equipment operability issues.

⑥ Reactor vessel head voiding phenomena that could occur in such a situation will be ignored, that is, the cooldown will not be stopped.

⑦ Battery power for instrumentation is assumed to last at least to the completion of the case run. This is necessary to allow operators to maintain heat removal by feeding and steaming the SGs. The key parameters to be monitored for this evolution are:

- Pressurizer Level
- Hot Leg Temperature
- Cold Leg Temperature
- Pressurizer Pressure
- Steam Generator Pressure
- Steam Generator Water Level

⑧ Instrument air supply for control devices or other means for manual operation will be available (TDAFW pump controls, AFW flow control, etc.).

⑨ Nominal SIT gas pressure and water volumes

⑩ All rods insertion for reactor trip with equilibrium xenon.

⑪ The potential for AFW source heatup during the long term will be addressed by using either a bounding temperature for that source or modeling the TDAFW pump heat addition to the CST.

⑫ The maximum RCP seal leakage is assumed.

⑬ A 75°F/hr cooldown (nominal) will commence two hours after the complete loss of AC power and will continue until SG pressure is equal to 120 psia.

⑭ Best estimate core physics data will be used.

⑮ Heat losses to containment ambient will be evaluated for the longer-term duration scenarios, although RCS mass loss may have more of an effect on pressure than will heat losses from the NSSS.

2.3 Win-NPA Simulation Code & Models

The Windows-based Nuclear Plant Analyzer (Win-NPA) [4] is an interactive simulation system to analyze the behavior of nuclear power plants during both normal and abnormal operational transients and to show the results with various graphical user interface displays through computer monitors. The Win-NPA has the following characteristics.

- Real-time, best-estimate simulation for NPP
- Accurate, easy-to-use & understand analysis tool
- Handy distribution package
- Dedicated tools for development & customization
- Supports external interfaces (DB, I&C devices)

The Win-NPA performs a simulation using the thermal hydraulic and control model based on the CENTS computer code [5] which is used for the baseline analysis of the CE type nuclear power plants in the "WCAP-17601-P".

The xenon reactivity feedback is modeled in the WIN-NPA as an increase or decrease in the scram rod worth depending on the time after trip.

The figure 1 shows the primary system nodes and flow-paths configuration of the Win-NPA.

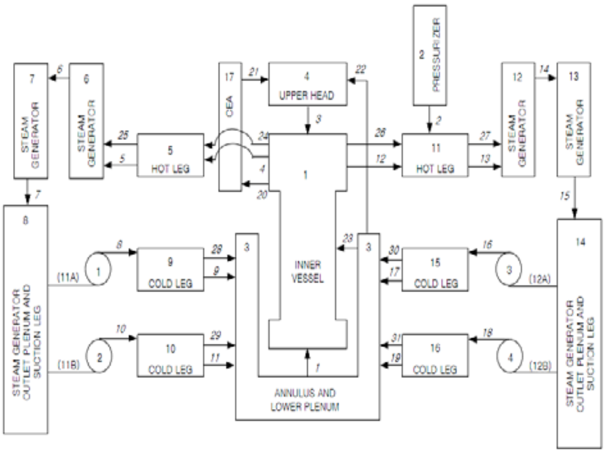


Figure 1. Node & Flow Path Configuration

2.4 Operator Actions for the Analysis

The following operator actions were assumed during an ELAP.

A. Operators take control of SG level ten minutes after the ELAP using only the TDAFW pump. Operators maintain ≥ 50% NR level in both SGs as best as possible. The advantage of maintaining a high SG liquid inventory is that, in the event of a temporary or permanent loss of auxiliary feedwater, additional heat removal can be supplied to avoid RCS heatup and pressurization.

B. Operators isolate CBO leakage for twenty minutes after event initiation (where modeled, 1 gpm of unidentified leakage is maintained).

C. Operators take control of one ADV on each SG two hours after the ELAP. Cool the Plant at a rate of 75°F/hr until SG pressure reaches 120 psia. SG pressure is maintained at 120 psia by closing the ADVs to keep a high steam pressure enough to prevent adverse impact to the TDAFW pump.

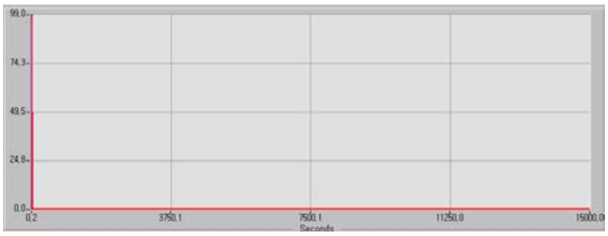
D. Operators heat the RCS at a rate of 75°F/hr to increase pressure and reduce SIT injection by closing the ADVs if pressurizer level goes above 40 ~ 50%.

2.5 Results of the Analysis

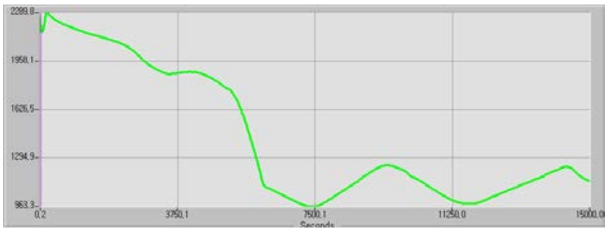
The Sequence of Events in the Table 1 provides more detail into the timing of various thermal-hydraulic events during the ELAP. In addition, Figures 2-1 through 2-6 provides time dependent behavior for key plant parameters.

Table 1: ELAP Sequence of Events for the OPR1000

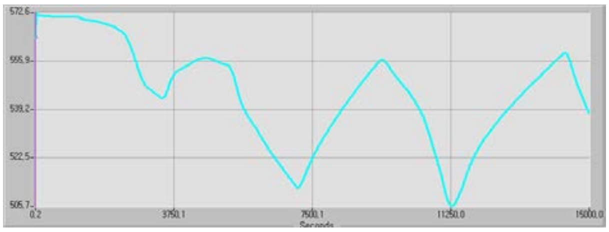
Time (sec)	Events	Comments
0.0	ELAP occurs causing the following: · Reactor Trip · Turbine Trip · Loss of Charging, Letdown · Loss of PZR Heaters · Loss of RCPs · Loss of MFW (coastdown in 5 sec) · Loss of Steam Dumps · Loss of RCP CBO Component Cooling Water · 15 gpm/pump; RCP seal leakage	
6.1	Main Steam Safety Valves Lift	1,264.7 psia setpoint (lowest setpoint valve)
600	Operator takes manual control of AFW	
1,090	AFW flow begins to both SGs	TDAFW pump only
3,600	Operator takes manual control of ADVs	
18,000	Operator commences cooldown to maintain ~50°F cold leg subcooling	
21,600	Operator ceases cooldown, maintains SG pressure at current value	RCS P ≈ 1150 psia, SG P ≈ 760 psia, Tcold ≈ 512°F
86,400	Loop natural circulation ceases, reflux boiling begins	
234,000	Core uncovery begins	



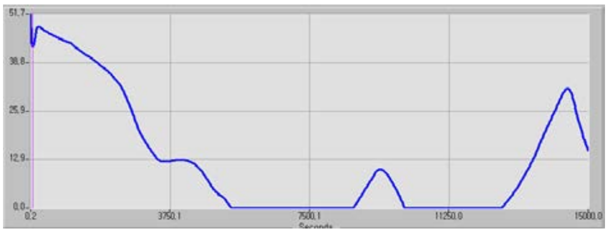
Figures 2. Core Power vs. Time



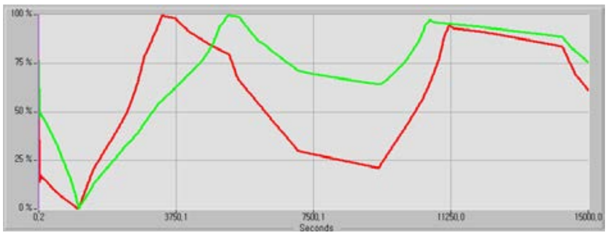
Figures 3. Pressurizer Pressure vs. Time



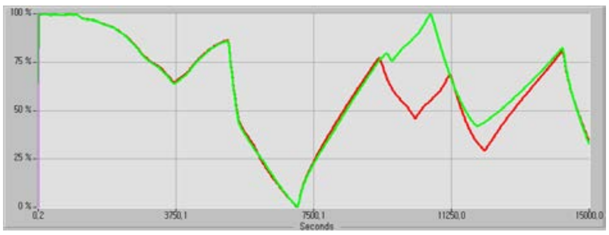
Figures 4. Cold-leg Temperature vs. Time



Figures 5. PZR water level vs. Time



Figures 6. SG water level (%WR) vs. Time



Figures 7. SG Pressure (%) vs. Time

2.6 Review of the Acceptance Criteria

2.6 .1 Preventing Core Damage

One acceptance criterion for the analyzed ELAP scenario is that no core damage will occur. Coping times will be calculated such that they preclude core damage.

2.6 .2 No Recriticality

There shall be no return to criticality once the loss of all AC power has occurred. To ensure that the plant remains subcritical, a limit of K_{eff} less than 0.99 is set. The exact needed level of subcriticality is somewhat subjective, but K_{eff} of 0.99 was chosen because it provides some margin to account for the best estimate or generic reactor physics parameters assumed in this analysis.

3. Conclusions

It is assumed for this case that sufficient SG secondary makeup inventory exists or can be attained, so that the duration of the ELAP prior to core damage is dependent solely upon the loss of inventory from the RCS. Even with a limited RCS cooldown and depressurization, and conservatively high assumed RCP seal leakage, the plant can be sustained for over 65 hours prior to core uncovery.

REFERENCES

[1] NRC, SECY-11-0093; Near-Term Report and Recommendations for agency Actions Following the Events in Japan, August 19, 2011.
[2] NRC, SECY-11-0137; Staff Requirements: Prioritization of Recommended Action to be taken in response to FUKUSHIMA Lessons Learned, December 15, 2011.
[3] PWROG, WCAP-17601-P, Rev.0; Reactor Coolant System Response to the Extended Loss of AC Power Event for Westinghouse, Combustion Engineering and Babcock & Wilcox NSSS Designs, August, 2012.
[4] KEPCO E&C, KOPEC/00 - TN – 0AF; Final Report of Development of the Best-Estimate Real Time Nuclear Plant Analyzer (NPA), December, 2000.
[5] WCAP-16248-P, Rev.0, User's Manual for the CENTS Code, March, 2004.
[6] CE Owner's Group, CEN-152, Rev. 5, Volume 2, Emergency Procedure Guidelines.
[7] Shin-Kori Unit 1, 2 Final Safety Analysis Report, KHNP.