Station Blackout and Nuclear Safety

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ABSTRACT

The safety of the nuclear power plant depends on the availability of the continuous and reliable source of electrical energy during all modes of operation of the plant. The station blackout corresponds to a total loss of all alternate current (AC) power as a result of complete failure of both offsite and onsite AC power sources. The electricity for the essential systems during station blackout is provided from the batteries installed in the nuclear power plant. The results of the probabilistic safety assessment show that station blackout is one of the main and frequently the dominant contributor to the core damage frequency. The accident in Fukushima Daiichi nuclear power plants demonstrates the vulnerability of the currently operating nuclear power plants during extended station blackout events.

The U.S. NRC Station Blackout Rule describes procedure for the assessment of the size and capacity of the batteries in the nuclear power plant. The description of the procedure with the application on the reference plant and identified deficiencies will be presented.

The implications of the extension of the station blackout coping capability on the safety of the nuclear power plant will be analysed with state-of-the-art probabilistic and deterministic methods applied on reference models of the nuclear power plants. Obtained results show large decrease of core damage frequency with the extension of the station blackout coping capability. The extension of blackout coping capability results in delay of the core heat up for at least the extension interval. Operation of the steam driven feedwater system results in no core heat up for 72 hours, even in the presence of the reactor coolant pumps leakage. The main conclusions from the analysis and recommendations considering prolonged blackout of the nuclear power plant are given.

1 INTRODUCTION

The main purpose of the nuclear safety is the prevention of the release of radioactive materials, ensuring that the operation of nuclear power plants (NPP) does not contribute significantly to individual and societal health risk. The main specific issue of the nuclear safety is the need for removing the decay heat, necessary even for a reactor in shutdown.

The nuclear power plant power systems are divided into safety related Class 1E and Non-1E power system. They are equipped with the continuous and reliable sources of electrical energy in order to sustain the effective cooling of the fuel. The emergency power sources of the Class 1E power system include on-site diesel generators and batteries.

The loss of offsite power (LOOP) initiating event occurs when all electrical power to the plant from external sources is lost. A total loss of all alternating current (AC) power as a result of complete failure of both offsite and onsite AC power sources is referred to as a station blackout (SBO).

The results of the Probabilistic Safety Assessment (PSA) show that initiating events LOOP and SBO are the most important contributors to the core damage frequency (CDF)
including the shutdown CDF [1]. Figure 1 shows the percentage share of the SBO and LOOP in the obtained CDF for four operational plants and two advance designs [2-4]. The SBO-S and SBO-L are designations for the SBO shorter and longer than design coping capability. During an extended SBO functional failure would occur for nearly all instrumentation and control systems leading ultimately to the core damage.

![Figure 1: Percentage contribution of LOOP and SBO to CDF](image)

The accidents in the Chernobyl [5] and Fukushima Daiichi NPP [6] support the PSA results. The accident in Unit 4 of the Chernobyl NPP started as operators initiated LOOP. The test that initiated the accident had to verify operation of the main feedwater pumps during a loss-of-coolant accidents and simultaneous LOOP. A prompt, supercritical reactor excursion led to the disastrous failure of the reactor.

On March 11, 2011, the Tohoku-Taiheiyou-Oki Earthquake occurred near the east coast of Honshu, Japan [7]. The earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Daiichi NPP as the result of a sustained loss of both the offsite and on-site power systems [6]. Units 1 through 3, which had been operating at the time of the earthquake, scrammed automatically. Following the loss of electric power to normal and emergency core cooling systems and the subsequent failure of back-up decay heat removal systems, water injection into the cores of all three reactors was compromised, and reactor water levels could not be maintained resulting in core damage. Efforts to restore power to emergency equipment have been hampered or impeded by damage to the surrounding areas due to the tsunami and earthquake [8] including common-cause failure (CCF) of NPP on-site distribution system due to the flooding and ground motion.

Both accidents support the importance of the AC power for the safety of NPP.

### 2 STATION BLACKOUT RULE

Based on concerns about SBO risk and associated reliability of emergency diesel generators, the U.S. Nuclear Regulatory Commission (NRC) issued the SBO rule [9] and the associated Regulatory Guide [10]. The SBO rule [9] requires that NPP must have the capability to withstand an SBO and maintain core cooling for a specified duration known as station blackout coping capability. The associated Regulatory Guide “Station Blackout” [10] provides procedure for assessment of the station blackout coping capability considering the factors identified in the SBO rule. The acceptable station blackout duration capability, in hours, is assessed from the table with one section given in Table 1.
Table 1: Acceptable SBO coping duration capability [hours]

<table>
<thead>
<tr>
<th>Offsite power design characteristics group</th>
<th>Emergency AC Power Configuration Group [C]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Average EDG reliability</td>
</tr>
<tr>
<td></td>
<td>0.975</td>
</tr>
<tr>
<td>P1</td>
<td>4</td>
</tr>
<tr>
<td>P2</td>
<td>4</td>
</tr>
<tr>
<td>P3</td>
<td>8</td>
</tr>
</tbody>
</table>

The emergency AC power configuration group is assessed from the number of the emergency AC power sources available at the site, including shared and swing generators. The Krško NPP is classified in AC Power configuration group [C]. The offsite power design characteristics group addresses expected frequency of loss of offsite power and expected restoration time considering independence of the offsite power sources and expected frequency of severe weather. The NPP Krško is categorized in [P1] offsite power group. Table 1 shows that acceptable blackout coping duration of 4 hours is obtained.

The common-cause failures of all onsite and offsite power resulting from naturally occurring external events, such as earthquakes and flooding, are not considered in the Regulatory Guide [10]. The SBO rule [9] does not require the ability to maintain reactor coolant system integrity (i.e., PWR reactor coolant pump seal integrity) or to cool the spent fuel pool. The SBO rule focuses on preventing fuel damage and therefore does not consider the probability for the buildup of hydrogen gas inside containment during a prolonged SBO condition and the potential need to power hydrogen igniters in certain containment designs to mitigate the buildup of hydrogen. In certain designs the hydrogen buildup may result in the need to vent the containment. The SBO rule does not require consideration of the impact on the station, and particularly on the onsite AC power system, of the natural event that caused the loss of offsite AC electrical power.

The NPP SBO coping capability can be improved with hardware modifications including installation of additional emergency diesel generators and/or additional batteries, improvement of the reliability of the current emergency diesel generators and establishing and improving the current SBO coping procedures. The results of the analysis of the impact of the hardware modifications on the safety and parameters of the NPP are presented in the following sections.

3 NPP MODELS

The PSA model of the Surry Unit 1 NPP is developed on the basis of the plant model and Level 1 PSA analysis given in the corresponding references [11]. The PSA model includes 18 Event Trees, 171 Fault Trees and 581 Basic Events. The PSA model considers the latest grid restoration times [2]. The crossties to the second unit electrical systems are removed from the model including the third swing generator available at the site.

Six PSA models of the NPP, including starting reference model, were developed and analyzed. The analyzed models include: 2EDG - reference model of the NPP with two EDG; 3EDG - NPP model with three EDG; 2EDGB - NPP model with two EDG and increased battery capacity; 3EDGB - NPP model with three EDG and increased battery capacity; 3CCF - NPP model with three EDG and increased CCF; 3CCFB - NPP model with three EDG and increased battery capacity and CCF.

The increased capacity of the batteries is considered SBO event tree with the extension of the blackout coping capability from 7 to 12 hours. The increased CCF probability is analyzed with twice larger CCF of all three EDG compared to the CCF in the reference model. Other
support systems, including necessary coolant and instrumentation air capacity, are assumed to be available in the analysed models.

The Relap5 input model of PWR nuclear power plant is used for the assessment of the nuclear power plant parameters [12]. The input model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. Included are all important components of reactor coolant system and secondary side, reactor protection system, control systems and safety systems, model of the steam generators and auxiliary feedwater system.

The following scenarios with or without Reactor coolant pumps (RCPs) seal leakage (1.32 l/s = 21 gpm) and with or without available turbine driven auxiliary feedwater system (TD AFWS) were analyzed by the Relap5 computer code:

- SBON – SBO without RCPs Seal Loca and TD AFWS available
- SBOS – SBO with RCPs Seal Loca and TD AFWS available
- SBOS7 – SBO with RCPs Seal Loca and TD AFWS available 7 hours
- SBOS12 – SBO with RCPs Seal Loca Seal Loca and TD AFWS available 12 hours

Obtained results from the analyzed scenarios are presented in the following sections.

4 PSA RESULTS

For the basic reference PSA model 2EDG, described in Section 3, CDF=2.52E-5 [/yr] is obtained. Figure 2 shows obtained decrease of the core damage frequency $\Delta$CDF compared to the basic model obtained for the analyzed models described in Section 3.

![Figure 2: Decrease $\Delta$CDF [%] compared to the CDF of reference PSA models](attachment:image.png)

Figure 2 shows that largest decrease is obtained for 3EDGB with additional EDG and increase of batteries capacity with comparable $\Delta$CDF obtained in 2EDGB with increased batteries capacity. The obtained $\Delta$CDF in 2EDGB is twice larger than the decrease in 3EDG obtained with the installation of the third EDG. Figure 2 shows the implications of the increased CCF of the EDG on the obtained $\Delta$CDF with smallest $\Delta$CDF obtained in 3CCF. The $\Delta$CDF for the 3CCFB is comparable to the results of 3EDGB model indicating small sensitivity of the model with increased batteries capacity to the CCF of the EDG.
5 RELAP5 RESULTS

The results obtained by the Relap5 computer code are given in the Figures 3 to 5 for 24 hours time window and Figure 6 for time 72 hours. In Figure 3 average core rod cladding temperature is shown. Figure 3 show fast progression of accident resulting in the core heat up and start of the core degradation when TD AFWS is not available in SBOS7 and SBOS12.

When TD AFWS is available the presence of a leak at the RCPs seals is important for the progression of the event. The leak presence causes decrease of the reactor coolant system (RCS) mass inventory. The other source of RCS mass inventory discharge is through the pressurizer safety valves when setpoint pressure is reached. In case of SBON the TD AFWS is not sufficient at first 21 hours to remove all decay heat. Figure 4 shows that the RCS pressure increases till the safety valve setpoint pressure with few short steam discharges in the period from 5 to 22 hours. Figure 5 show that the RCS mass inventory loss is small in SBON. Figure 4 shows that in the presence of RCP seal leakage the failure of the TD AFWS in SBOS7 and SBOS12 results in fast increase of the RCS pressure. Figure 5 show that the consequent opening of the pressurizer safety valves causes significant coolant inventory loss. The cooling of the RCS could be significantly increased, when depressurization would be done by operators through the secondary side relief valves. In this study it was assumed that without DC power the main steam relief valves are not operable and discharge is available through steam generator safety valves. Comparison of the results for the scenarios SBOS7 and SBOS12 on Figure 3 and Figure 5 show that extension of the station blackout coping capability for 5 hours results in delay of the core heatup for at least the extension time. Figure 4 show that the increase of the reactor coolant system pressure is faster in SBOS12 compared to the SBOS7 after the failure of the TD AFWS. This is resulting from the smaller RCS coolant inventory (decreased heat transfer from core to the steam generators) when TD AFWS stops operating in the SBOS12 compared to the SBOS7.

Figure 6 shows that in case of the TD AFWS normal operation the core will maintain covered with coolant both in absence and in presence of the RCP seal leakage. The operators have this time window for the restoration of AC power from either on-site or offsite sources.

![Figure 3: Average fuel cladding temperature at the top of the core](image-url)
Figure 4: Pressure of the reactor coolant system

Figure 5: Mass of the reactor coolant system

Figure 6: Mass of RCS (left axis) and average fuel cladding temperature (right axis)
6 DISCUSSION

In response to the accident at the Fukushima Daiichi NPP, the U.S. NRC prepared report with the review of insights from the accident and list of recommendations in order to improve reactor safety [13]. The report recommends actions to ensure protection from the external events, enhance mitigation and strengthen emergency preparedness.

The report recommends strengthening of the SBO mitigation capability of all operating and new reactors for design-basis and beyond-design-basis external events and initiation of rulemaking for revision of the SBO Rule [9]. The SBO enhancement recommendations include: minimum of 8 hours blackout coping time, establishment of equipment, procedures, and training necessary to implement a station blackout coping time of 72 hours and assurance of the availability of the offsite resources to support extended NPP blackout.

The 8-hour coping systems and equipment should be protected from damage from all design-basis events and extended beyond-design-basis events by either locating the equipment one level (i.e., 5 to 6 meters) above the plant design-basis flooding level or in water-tight enclosures, ensuring that core and spent fuel pool cooling is maintained and unmanageable leakage of coolant does not occur (e.g., from PWR reactor coolant pump seal failure). The 8-hour coping capability should only rely on permanently installed equipment.

In the report the assessment of the power grid and on-site distribution system reliability [14-16], is not considered. Data analysis [17] from year 1986 through year 2004 reveals that SBO risk was low when evaluated on an average annual basis due to the plant modifications in response to the SBO rule [18]. When focus is on grid-related LOOP events, the SBO risk has increased, with the largest increase in frequency and length notified for weather related LOOP. Considering the creation of internal electricity energy market, increase of the share of the renewable sources in the grid and increased cross border exchange of electricity, the number and length of grid blackouts is expected to increase [19].

7 CONCLUSIONS

The results of the probabilistic safety assessment support the importance of the station blackout initiating event and availability of the electrical energy for the nuclear safety of the nuclear power plants.

Description of the procedure for the assessment of the station blackout coping capability with the application on the reference plant and identified deficiencies is presented.

Implications of plant modifications improving the station blackout coping capability on the core damage frequency the reference plant are analyzed with the probabilistic safety assessment. The obtained results show that largest decrease of core damage frequency is obtained with the extension of the station blackout coping capability. The obtained results show the importance of the common cause failures of the emergency diesel generators for the overall safety of the plant.

The Relap5 results show that extension of the station blackout coping capability results in delay of the core heatup for at least the extended coping time. The results show that effective and long term cooling of the reactor can be sustained with the auxiliary turbine driven feedwater system under assumption of the availability of other auxiliary systems. In presence of the seal leakage with loss of 1.32 l/s, the plant can survive initial 72 hours if auxiliary turbine driven feedwater system is available and operational.

The latest recommendations and expected revisions to the corresponding regulatory requirement by the U. S. Regulatory Commission are presented in last section. The proposed modifications and recommendations are in line with the obtained results in this study.
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REFERENCES