Integrated Safety Assessment of the Steam Generator Tube Rupture Accident in a PWR using Best Estimate Codes

G. Jimenez
1Nuclear Engineering Department, Universidad Politécnica de Madrid
Madrid, Spain
gonzalo.jimenez@upm.es

M.J. Rebollo1, C. Queral1, J. Gil2, I. Fernández2, J. J. Gómez-Magán2, M. Sánchez-Perea3, J. Hortal3, Jm. Izquierdo3, E. Meléndez3
2Indizen Technologies S.L., Madrid, Spain
3Consejo de Seguridad Nuclear (CSN), Madrid, Spain

ABSTRACT

Steam Generator Tube Rupture accidents in Pressurized Water Reactors are known to one of the most demanding transients for the operating crew. SGTR are special transient as they could lead to radiological releases without core damage or containment failure, as they can constitute a direct path to the environment.

The SGTR is analysed from a Deterministic and Probabilistic point of view in the Safety Analysis, although the assumptions of the different approaches regarding the operator actions are quite different.

In the beginning, the way of analysing the SGTR within the Deterministic Safety Analysis was not crediting the operator action for the first 30 min of the transient, assuming that the operating crew was able to stop the primary to secondary leakage within that time. However, the different real SGTR accident cases happened in the USA and over the world demonstrated that can operators took more than 30 min to stop the leakage in real life. Some methodologies took were raised to cover that issue.

From the probabilistic point of view, the operator actions are taken into account to fix the headers in the event tree. The available times are used to establish the success criteria for the headers. However, in such a dynamic sequence as SGTR, the operator actions are very dependent on the time available left by the other human actions, so the appropriate way of analysing that kind of sequences could be though a Dynamic Event Tree

This paper presents on one hand the results of comparing those methodologies included in the different Deterministic Safety Analysis with a PWR Westinghouse three loop model in TRACE code (Almaraz NPP) with best estimate assumptions but including deterministic hypothesis such as single failure criteria or loss of offsite power. The behaviour of the reactor is quite diverse depending on the different assumptions made in each methodology. On the other hand, the high conservatism included in the radiological hypothesis makes that at the end all the results are quite far from the regulatory limits.

On the other hand, the Integrated Safety Assessment (ISA) methodology, developed by the Spanish Nuclear Safety Council (CSN), has been applied to a thermo-hydraulical analysis of a Westinghouse 3-loop PWR plant by means of the dynamic event trees (DET). The ISA methodology allows obtaining the SGTR Dynamic Event Tree taking into account the operator actuation times. Simulations are performed with SCAIS (Simulation Code system for Integrated Safety Assessment), which includes a dynamic coupling with MAAP thermal
hydraulic code. The results show the capability of the ISA methodology and SCAIS platform to obtain the DET of complex sequences.

1 INTRODUCTION

A Steam Generator Tube Rupture (SGTR) in a Pressurized Water Reactor (PWR) can lead to an atmospheric release bypassing the containment via the secondary system and exiting though the Pressurized Operating Relief Valves of the affected Steam Generator. That is why SGTR historically have been treated in a special way in the different Deterministic Safety Analysis (DSA), focusing on the radioactive release more than the possibility of core damage, as it is done in the other Loss of Coolant Accidents (LOCAs).

Main human actions within Emergency Operating Procedures (EOP) needed in order to optimally recover the transient are depicted in Fig. 1. With these actions in mind, the principal stages of SGTR sequences are:

- Reactor trip and Safety Injection (SI) signal.
- Identification and Isolation of the ruptured SG.
- Cooldown of the RCS system by means of the intact SGs.
- Depressurization of RCS to restore inventory.
- Terminate SI.
- Long term cooling.

Although it has been deeply study, the establishment of the hypothesis for the DSA of the SGTR is not an easy question, due to the complexity of the transient and the necessary operator actuation. At the beginning, in the NRC’s Standart Review Plan the rule of the 30 minutes without operators action, developed for LOCA conditions, were applied to the SGTR DSA too, see [1]. That hypothesis were based on the idea that the operators were capable of finishing the leakage of the ruptured tube within 30 minutes, so the most conservative assumption was to do it in minute number 30.

Nevertheless, the real SGTR events demonstrated the difficulty of finishing the leakage within those 30 minutes. None of the operator’s crew involved in the SGTR from 1975 to 1996 did it, see [2]. After SGTR accident of Ginna NPP happened in 1982, a subgroup of PWR owners with Westinghouse Electric Company worked together to develop a methodology to analyze the SGTR accident taking into account the operating experience. That methodology was described in the WCAP-10698 that is called “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill”, see [3], which describes the way of doing the overfill calculation and its supplement “Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident”, [4], which describes the way of calculating the offsite doses from SGTR calculation. Both calculations have the single failure criteria: for the offsite dose calculation it normally is that the damaged PORV is stuck open for some minutes after the SG is isolated. For example, in the calculation of Watts Bar NPP, the PORV was determined to stuck open for 11 min, see [6].

There are one other alternative to WOG proposed methodology found in the public literature, the methodology developed by Kansas Gas & Electric Company, owners of Wolf Creek and Callaway, the SNUPPS (Standardized Nuclear Unit Power Plant System) plants. In that methodology, as it can be seen in WCAP-16265, [5], there is no overfill calculation, there are two different offsite dose calculations: the first one includes the failure of the faulted SG PORV within the SCRAM for 20 min and the second one the overfill of the faulted SG due to the AFW malfunction, failing the faulted SG PORV if there is liquid releases.
In the first part of this paper, the methodologies with operator action (WOG and SNUPPS proposed methodologies) are compared with the classical FSAR methodology of no operator action for the first 30 min with a common best estimate model (Almaraz NPP TRACE model) and hypothesis (normal operating conditions with LOOP at SCRAM), to evaluate the differences in terms of offsite dose.

From the probabilistic point of view, the operator actions are taken into account to fix the headers in the event tree. The available times are used to establish the success criteria for the headers. However, in such a dynamic sequence as SGTR, the operator actions are very dependent on the time available left by the other human actions, so the appropriate way of analysing that kind of sequences could be though a Dynamic Event Tree.

As part of the collaboration between Universidad Politécnica de Madrid (UPM), Indizen Technologies and the Spanish Nuclear Safety Council (CSN), an analysis of Steam Generator Tube Rupture (SGTR) sequences in a PWR Westinghouse has been performed with SCAIS, see [7]. The objective of the analysis has been the application of the Integrated Safety Assessment (ISA) methodology in order to obtain the SGTR Dynamic Event Tree taking into account the operator actuation times.

The ISA methodology has been developed by the Modeling and Simulation (MOSI) branch of CSN. ISA is an adequate method to perform the uncertainty analysis, especially suited to compute uncertainties for those sequences where some of the events occur at uncertain times (time delay of operator response and other stochastic events) along with usual parametric uncertainties. The numerical results of this methodology include among others the Damage Exceedance Frequency (DEF) for the sequences stemmed from an initiating event. This is done along with the delineation of the dynamic event tree and the identification of the damage domain (DD) of the sequences that contribute to the total DEF. The damage domain is defined as the region of the space of uncertain parameters of interest that results in damage (see [8]).

In the second part of this paper the ISA methodology is applied to a SGTR accident, evaluating the capability of this methodology to cope with such a difficult transient.

2 RESULTS FROM THE SINGLE FAILURE CRITERIA CASES WITH ALMARAZ NPP TRACE MODEL

Almaraz NPP has two PWR units, it is located in Cáceres (Spain) and it is owned by a consortium of three Spanish utilities: Iberdrola (53%), Endesa (36%) and Gas Natural Fenosa (11%). The commercial operation started in April 1981 (Unit I) and in September 1983 (Unit II). Each unit is a three loop PWR Westinghouse. The nominal power is 2739 MWt and 977 MWe, respectively. The original Westinghouse steam generators were replaced between 1996 and 1997 and since then it is equipped with three Siemens KWU 61W/D3 steam generators. Reactor coolant pumps are single stage centrifugal model, designed by Westinghouse. The AFWS consists of one turbine driven pump and two motor driven pumps.

Almaraz I NPP TRACE model has 255 thermal-hydraulic components (2 VESSEL, 73 PIPE, 43 TEE, 54 VALVE, 3 PUMP, 12 FILL, 33 BREAK, 32 HEAT STRUCTURE and 3 POWER component), 740 SIGNAL VARIABLES, 1671 CONTROL BLOCKS and 58 TRIPS, Figure 1. This model has been validated with steady and transient conditions and verified with a large set of transients, see [9] to [16].
The different methodologies presented in the introduction were compared:

- No operator action for the first 30 min (classical FSAR methodology)
- SGTR with operator action, single failure is the damaged SG PORV stuck open at SG isolation time (first proposed methodology). Several times were tested to see the sensitivity to the isolation time (from 5 min to 40 min from the start of the transient) fixing the time that the valve is stuck open (11 min).
- SGTR with operator action, single failure is the damaged SG PORV stuck open 20 min at SCRAM (second proposed methodology)

The dose resulting from the classical methodology of no operator action for the first 30 min results in 3.84E-02 Sv for the boundary of the exclusion area (EAB) and 1.38E-02 Sv for the low population zone (LPZ) in the most limiting case, which was thyroid dose for Coincident Iodine Spike. The limit from RG 1.195 for SGTR accident in the Coincident Iodine Spike is 0.3 Sv, see [17], so the results are about 13% (EAB) and 4% (LPZ) of the regulatory limit.

The doses of the methodologies that includes operator actions have been compared to the methodology that don’t have them in their most limiting cases (thyroid dose for Coincident Iodine Spike) and are summarized in Table 1:

Table 1: Dose results from methodologies with operator action compared with the no operator action for 30 min case

<table>
<thead>
<tr>
<th>Methodology</th>
<th>Thyroid dose compare to no operator action case (Coincident Iodine Spike)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>First proposed methodology</td>
</tr>
<tr>
<td></td>
<td>Isolation /stuck open</td>
</tr>
<tr>
<td>11 min</td>
<td></td>
</tr>
<tr>
<td>Second proposed methodology</td>
<td></td>
</tr>
</tbody>
</table>

3 RESULTS OF THE DYNAMIC EVENT TREE SIMULATION

The objective of Block A of the ISA methodology is to simulate the dynamic event tree (DET) stemming from an initiating event. At present, the simulations of DET performed by coupling MAAP and DENDROS are performed in an automatic way, see [18].
In this study a SGTR event at full power in a three-loop PWR Westinghouse design has been considered. Such accidents begin as a breach barrier between the primary Reactor Coolant System (RCS) and the secondary side of the steam generator (SG), and provide a direct release path for RCS fluid to the environment via the secondary side (steam-dump, safety and relief) valves.

In a first step, several SGTR event trees (ET), corresponding to PSA studies of similar nuclear power plants (Westinghouse design with 3 loops) have been analyzed to build a generic SGTR event tree including EOPs (Generic Event Tree, GET). Sequence headers and success criteria considered after this review are:

SCRAM: Reactor trip; HPI: High Pressure Injection system (1/2 pumps); AFW: Auxiliary Feed Water system (1/3 pumps); ISO-SG: Isolate ruptured SG (close MSIV on ruptured SG); DSP-EP: Cooldown and Depressurization of RCS (cooldown dumping steam from PORVs on intact SGs and depressurization with pressurizer PORV); DSP-LP: Cooldown and depressurization of RCS (cooldown, 55 K/h rate, dumping steam from PORVs on intact SGs and depressurization with pressurizer sprays; FB: Feed and Bleed (1/2 HPI pumps, 1 PORV); C-PORV/RSIS: Terminate SI (control charging flow); REC-HP: High Pressure Recirculation (injection to 2/3 legs of RCS with 1/2 HPSI pumps and 1/2 LPSI pumps); RWST: Refill of Water Storage Tank; RHR: Residual Heat Removal. One of the main difficulties in SGTR simulations is the selection of the operator action timing, which is very plant-specific. Nevertheless the timing are in the same range comparing similar plants based on operator training on full-scope simulators, see [19] to [21]. Taking into account this information, a set of representative values for the DET-SGTR analysis has been obtained.

All simulations performed in this analysis with DENDROS-MAAP include as hypothesis Success criteria for headers SCRAM, AFW and RC-HP; in addition RWST and RHR headers are not included in the analysis. Therefore, the headers considered in simulations are: HPI (named H in DET), ISO-SG (named I in the DET), DSP-EP (named SD in DET), DSP-LP (named LD in DET), C-PORV/RSIS (named R in DET). Simulations are finished when RHR conditions are reached (success sequence), PCT > 1477.15 K (damage sequence) or Time > 24 hours. In addition, twenty four operator actions, corresponding to EOPs E-3, ES-3.1, ECA-3.1 and ECA-3.2, have been included in DET simulation.
The DET, Figure 2, provides the actuation time for systems and branching time for headers. For the purpose of DET unfolding, simulations do not consider time delays: each header occurs at the stimulus time or never. The identification of each sequence is carried out by the concatenation of header status: a header in upper case means success upon demand and lower case means failed upon demand. Main values of variables and times of events obtained from the sequences of DET are shown in Table 2.
Table 2: Sequence information obtained from the reference DET

<table>
<thead>
<tr>
<th>Sequence</th>
<th>RCS Press. Min. (Pa)</th>
<th>RCS Level Min. (m)</th>
<th>Intg. Mass Flowbreak (kg)</th>
<th>End Release Time (s)</th>
<th>Total Mass Steam Release (kg)</th>
<th>Max. PCT (K) [Time (s)]</th>
<th>ACCs Injec. Time (s)</th>
<th>RHR Time (s)</th>
<th>End RWST Time (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>DM0: H-I-SD-R</td>
<td>2.1E6</td>
<td>20.8</td>
<td>4.1E4</td>
<td>2378</td>
<td>3.1E5</td>
<td>603</td>
<td>-----</td>
<td>12053</td>
<td>-----</td>
</tr>
<tr>
<td>DM1: H-I-SD-r</td>
<td>2.1E6</td>
<td>20.8</td>
<td>4.3E4</td>
<td>2670</td>
<td>3.2E5</td>
<td>603</td>
<td>-----</td>
<td>-----</td>
<td>-----</td>
</tr>
<tr>
<td>DM2: H-I-sd-LD-R</td>
<td>2.5E6</td>
<td>20.8</td>
<td>5.0E4</td>
<td>2803</td>
<td>3.1E5</td>
<td>603</td>
<td>-----</td>
<td>12067</td>
<td>-----</td>
</tr>
<tr>
<td>DM3: H-I-sd-LD-r</td>
<td>6.7E6</td>
<td>20.8</td>
<td>1.4E5</td>
<td>9134</td>
<td>3.3E5</td>
<td>603</td>
<td>-----</td>
<td>----</td>
<td>67509</td>
</tr>
<tr>
<td>DM4: H-I-sd-Id</td>
<td>7.8E6</td>
<td>20.8</td>
<td>1.5E6</td>
<td>76811</td>
<td>4.0E5</td>
<td>603</td>
<td>-----</td>
<td>----</td>
<td>74264</td>
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<tr>
<td>DM5: H-I-LD-R</td>
<td>9.1E5</td>
<td>20.8</td>
<td>9.1E4</td>
<td>9981</td>
<td>4.4E5</td>
<td>603</td>
<td>-----</td>
<td>9981</td>
<td>-----</td>
</tr>
<tr>
<td>DM6: H-I-LD-r</td>
<td>2.5E6</td>
<td>20.8</td>
<td>1.4E6</td>
<td>54576</td>
<td>5.1E5</td>
<td>603</td>
<td>-----</td>
<td>----</td>
<td>53111</td>
</tr>
<tr>
<td>DM7: H-I-ld</td>
<td>7.4E6</td>
<td>0.57</td>
<td>1.6E6</td>
<td>-----</td>
<td>1.9E6</td>
<td>&gt;1500</td>
<td>-----</td>
<td>73215</td>
<td>-----</td>
</tr>
<tr>
<td>DM8: h-I-SD</td>
<td>1.1E6</td>
<td>20.8</td>
<td>1.4E4</td>
<td>949</td>
<td>3.1E5</td>
<td>603</td>
<td>-----</td>
<td>10380</td>
<td>-----</td>
</tr>
<tr>
<td>DM9: h-I-sd-LD</td>
<td>1.1E6</td>
<td>20.8</td>
<td>1.3E4</td>
<td>847</td>
<td>3.1E5</td>
<td>603</td>
<td>5151</td>
<td>11513</td>
<td>-----</td>
</tr>
<tr>
<td>DM10: h-I-sd-Id</td>
<td>7.8E6</td>
<td>20.8</td>
<td>1.3E4</td>
<td>847</td>
<td>3.1E5</td>
<td>603</td>
<td>-----</td>
<td>10017</td>
<td>-----</td>
</tr>
<tr>
<td>DM11: h-i-LD</td>
<td>9.1E5</td>
<td>20.8</td>
<td>5.2E4</td>
<td>-----</td>
<td>4.5E5</td>
<td>603</td>
<td>-----</td>
<td>----</td>
<td>-----</td>
</tr>
<tr>
<td>DM12: h-i-ld</td>
<td>6.9E6</td>
<td>0.57</td>
<td>1.5E5</td>
<td>-----</td>
<td>1.4E6</td>
<td>&gt;1500</td>
<td>47023</td>
<td>-----</td>
<td>-----</td>
</tr>
</tbody>
</table>

4 CONCLUSIONS

It can be seen in that the results are quite broad depending on the isolation time, Table 1. If the operating crew isolates the SG in less than 15 min, which is a reasonable time, the offsite dose are approximately the same as for the no operator action case. The worst case, 40 min for isolating with 11 min of release form the SG PORV is only 1.83 times the dose of the no operator action case (24% of the limit), being quite improbable times for a real SGTR accident. It can be concluded that the hypothesis form all the methodologies are quite conservative, as the offsite dose results stay quite far from the regulatory limit, being the worst case 24% of the limit.

Results of DET, Table 2, show that damage end status is reached for sequences DM7 and DM12 (Fig. 3) which corresponds to sequences S19 and S31 in GET, which were damage sequences too. However in the GET there is another damage sequence (S12) which corresponds to one success sequence DM4, in DET. It can be concluded that it is difficult to determine exactly the end states in such difficult transients as SGTR and the DET can be an useful tool to contrast the Generic Event Tree.
REFERENCES


