Simulación del accidente de SGTR en un PWR-W con TRACE para distintas metodologías de Análisis Determinista de Seguridad.

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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, the methodology developed by Westinghouse based on the NRC´s Standart Review Plan, included 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 this paper, the methodologies with operator action (WOG and KGEC 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.

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 2947 MWt and 1055 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].

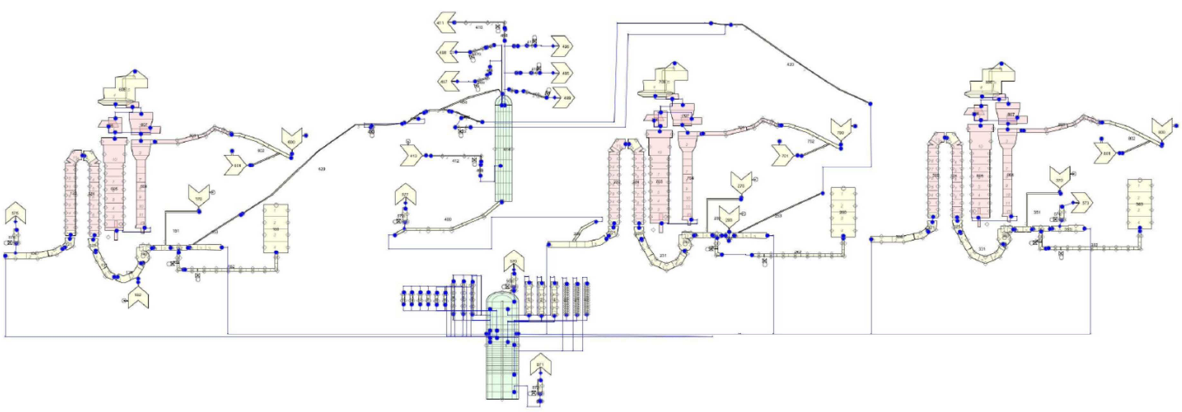

Figure 1 Simplified scheme of the Almaraz NPP TRACE model

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 limit of RG 1.195 in their most limiting case for dose (thyroid dose for Coincident Iodine Spike at the Exclusion Area Boundary), Table 1.

Methodology	Thyrod dose compare to the RG 1.195 limit (0.3 Sv) for Coincident Iodine Spike at the Exclusion Area Boundary							
First proposed methodology*								
Isolation /stuck open	5 min	10 min	15 min	20 min	25 min	30 min	35 min	40 min
11 min	4.16%	6.6%	9.93%	14.39%	19.30%	25.33%	32.33%	39.44%
Second proposed methodology*	13.02%							
No operator action for 30 min	12.82%							

Table 1. Dose results from methodologies with operator action compared to the RG 1.195 limit

* First proposed methodology: SGTR with operator action from the beginning, with LOOP. The single failure is the stuck open damaged SG PORV at SG isolation.
** Second proposed methodology: SGTR with operator action from the beginning, with LOOP. The single failure is the stuck open damaged SG PORV at SCRAM with 20 minutes to close the faulted PORV.

3. 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 15 min, the offsite dose is approximately the same as for the no operator action case. If the isolation time is less than 15, the dose is very reduced. The worst case, 40 min for isolating with 11 min of release form the SG PORV represent the 40% of the limit, so it is still far enough to a relevant offsite dose, 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 40% of the limit.

4. References

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