STEAM GENERATOR TUBE RUPTURE IN NUCLEAR POWER STATIONS

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ABSTRACT

A developed model for the U-tube steam generator performance in pressurized water reactor nuclear power station is introduced. The total loss of feed water flow is assumed as an initiating event in the mathematical model. The model is solved for different water levels in the steam generator secondary side. The minimum value of the water level is defined as a limiting level for the beginning of the rupture of some of the U-tubes of the steam generator.

The scenario of the accident consequences of the U-tube steam generator rupture is presented. Recommendation and learned lessons based on the model results are studied.

Keywords **:** Steam Generators , Tube Rupture , Loss of Feedwater , Anticipated Transient without Scram .

INTRODUCTION

The anticipated transient without scram (ATWS) total loss of main feedwater (TLMFW) for the U-tube steam generator (UTSG) in a pressurized water reactor (PWR), is the main contributor for the rupture of the tubes of the UTSG [1-3]. Several models have been developed for the simulation of the transients in the nuclear power plants and analyzing the behavior of the UTSG, which include natural circulation or expert diagnosis.

A best-estimate RELAPS model of the AP600 nuclear power plant [4] has been developed to evaluate the capability of RELAP5 to simulate the integrated behavior of the innovative passive safety features of this plant [1-7].

A Simulation model consists of a set of coupled linear differential equations for the PWR nuclear power plant [5] was developed to study the performance of the UTSG under accidental conditions in the reactor core. The model includes the reactor core, pressurizer, primary system piping, and a U-tube recirculation type steam generator. The model was set up to simulate a spectrum of transients and accidents as wide as possible by including the complete plant analysis.

In this paper, the UTSG behavior in a pressurized water nuclear power station was analyzed during the event of the ATWS total loss of main feedwater flow. The key parameters responsible about the partial rupture of the UTSG were analyzed. Also, the minimum value of the water level in the steam generator shell side was defined. The scenario of the event consequences leading to the tube rupture and impacts are analyzed.

MATHEMATICAL MODEL

The Primary Coolant Loop Model

The only variable of interest in the primary loop is the variation in the primary loop water temperature from its steady-state values. Previous studies have shown that for load-following analysis, a three-lump energy balance model using the reactor outlet temperature as the input to the first lump gives an adequate representation of the primary loop dynamics. The three primary loop lumps are:-

- The primary coolant volume between the reactor core outlet and the steam generator U-tube inlet.

The primary coolant volume in the steam generator U-tubes

- The primary coolant volume between the steam generator outlet and the reactor core inlet.

It is assumed that the primary coolant volume in the steam generator system operates at constant pressure e.g. 2250 psia , with a constant coolant mass flow rate, and that the specific heat of the coolant can be represented by an appropriate average value with these assumptions, the three primary loop energy balance equations are as follows:

Lump 1 : Core upper plenum, hot leg, and steam generator inlet plenum:

$$\frac{d}{dt} T_{pi}(t) = \frac{W_p}{M_h}[T_{r0}(t) - T_{pi}(t)] \qquad (1)$$

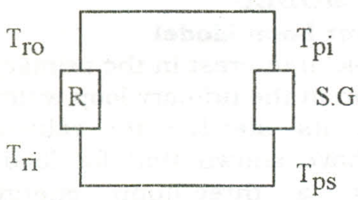
Lump 2 : Steam generator U-tube coolant volume:

primary water energy balance

$$\frac{d}{dt} T_{ps}(t) = \frac{W_p}{M_u}\left[T_{pi}(t) - T_{ps}(t)\right] - \frac{U_{pm}S_{pm}}{M_u C_p}\left[T_{ps}(t) - T_m(t)\right] \qquad (2)$$

Lump 3 : Steam generator outlet plenum, cold leg, and core inlet plenum:

$$\frac{d}{dt} T_{ri}(t) = \frac{W_p}{M_c}\left[T_{ps}(t) - T_{ri}(t)\right] \qquad (3)$$



Where :

T_{pi}, T_{ps} = SG primary coolant inlet and outlet temperatures.

T_{ri}, T_{ro} = Reactor coolant inlet and outlet temperatures.

T_m = SG tube metal temperatures.

W_p = Primary coolant mass flow rate.

M_h = Coolant mass in hot leg.

M_u = Coolant mass in U-tubes.

M_u = Coolant mass in the cold leg.

C_{pl} = Specific heat in primary coolant.

U_{pm} = Heat transfer coefficient between metal tube and primary side.

S_{pm} = Surface area of U-tubes in the primary side.

Steam Generator Model

In the steam generator, the primary coolant pump in the U-tubes transmits heat through the U-tube metal to the water on the secondary side. The variables of interest are the primary water temperature (Equation 2) and the secondary-side steam pressure and water temperature. As in the primary loop, lumped parameter balance equations representing the steam generator dynamics are:

U-tube metal lump:

U-tube metal energy balance

$$\frac{d}{dt} T_m(t) = \frac{U_{pm} S_{pm}}{M_m C_m}\left[T_{ps}(t) - T_m(t)\right] - \frac{U_{ms} S_{ms}}{M_m C_m}\left[T_m(t) - T_s(t)\right] \qquad (4)$$

Secondary water lump:

$$K_{ps}\frac{d}{dt}P_s(t) = U_{ms}S_{ms}\left[T_m(t) - T_s(t)\right] + W_s(t)\left[-h_g + cp_2 T_{FW}(t)\right] \qquad (5)$$

where $K_{ps} = M_{sw}\frac{\partial h_f}{\partial P} + M_{ss}\frac{\partial h_g}{\partial P} - M_{ss}\frac{h_{fg}}{v_{fg}}\frac{\partial v_{fg}}{\partial p}$

and

T_s = Secondary coolant temperature.

T_{FW} = Feedwater temperature.

M_m = Mass of U-tubes metal.

C_m = Specific heat of U-tubes metal.

C_{p2} = Specific heat of feedwater.

W_s = Secondary water mass flow rate.

P_s = Steam pressure of secondary side.

U_{ms} = Heat transfer coefficient between metal tube and secondary side.

S_{ms} = Surface area of metal U-tubes in the secondary side.

M_{sw} = Mass of saturated water.

M_{ss} = Mass of saturated steam.

V_{fg} = Specific volume of fluid and gas mixture.

h_g, h_{fg} and h_f = Secondary side saturated water enthalpies for vapor, evaporation, and liquid at the steam generator operating conditions .

In the above, h_g, h_{fg} and h_f are the secondary side saturated water enthalpies for vapor, evaporation, and liquid at the steam generator operating conditions.

$$\text{if } \eta = \frac{\text{surface area of the tubes immersed } (A_w)}{\text{total surface area of the tubes } (A_t)}$$

$$\therefore \eta = \frac{A_w}{A_t} = \frac{Z_w}{Z_t} \qquad (6)$$

where :

$A_w = \pi \; d_t n \; Z_w$

$A_t = \pi \; d_t n \; Z_t$

Z_w = water level in the SG secondary side,
Z_t = SG tube height,
d_t = tube diameter,
n = number of tubes

Usually Z_w is measured and controlled by special controller , and also it represents an indication to the efficiency of the feedwater system.

The water level, Z_w , can be calculated from:

$$M_{sf} V_{sf} = \frac{\pi}{4} D^2 Z_w - V_{UT}, \qquad (7)$$

where :
D= diameter of the SG,
M_{sf} = mass of saturated fluid
V_{sf} = specific volume of saturated fluid
V_{UT} = volume of U tubes.

The SG water level is controlled by a three element feedwater flow controller, which maintains a programmed water level as a function of turbine load. The three element feedwater controller regulates the feedwater valve by continuously comparing the feedwater flow signal, water level signal, the programmed level and the pressure compensated steam flow signal. we can conclude that for safety consideration, Z_w must be greater than Z_t, and the minimum level must be equal to Z_t.

PWR loss of feedwater

The ATWS Complete loss of main feedwater (LOFW) has been widely determined to produce the most severe RCS overpressurization in worst rupture of the steam generator tubes. The details of the PWR response to an ATWS LOFW are dependent upon many PWR design parameters. These parameters include core power density, RCS volume, pressurizer volume, main steam system design, safety injection system design, shutdown cooling system design and containment spray system design.

The major features of the ATWS LOFW

In the large RCS overpressurization with a relatively slow time response the ATWS LOFW is initiated when main feedwater flow is terminated to all steam generators . The LOFW produces SG pressure and temperature increases and subsequently, a reduction of the heat removal from RCS to the SG secondary water. This reduction of heat removal from the RCS causes the reactor coolant temperature to increase. Increased coolant temperature causes a reduction in core power due to moderator reactivity feedack.

Thermal expansion of the reactor coolant causes an insurge of subcooled primary coolant into the pressurizer, thus increasing pressurizer pressure and water level. Increased RCS pressure produces activation of the pressurizer sprays, however, then flow rate is insufficient to terminate the RCS pressure rise .

The RCS pressure rises sufficiently to open the power operated relief valves (PORVS) on the RCS . The PORVS then stabilize the RCS pressure while the heat transfer rate from the RCS continues to decrease , the reactor coolant temperature continues to increase, and the pressurizer water level continues to rise.

Sequence of event during total loss of normal feedwater with failure of reactor protection system :

1. Loss of all main feedwater form
2. Low SG level reactor trip signal
3. High Pressurizer pressure reactor trip signal
4. Pressurizer power operated relief valves open
5. Auxiliary feedwater form begins
6. Bypass- to - condenser valves open
7. Pressure reactor trip signal

8. Pressurizer fills
9. Minimum SG secondary liquid inventory
10. Maximum RCS pressure
11. Maximum RCS average temperature
12 Saturation reached in hotted reactor coolant
13. Reactor coolant pump cavitation
14. Pressurizer steam bubble reforms
15. Pressurizer safety valves close
16. Pressurizer power operated relief valves close
17. Operator manually initiates soluble poison injection
18. Reactor coolant system pressure and temperature decays with power decreases

SG tube Rupture Accident

A SG tube rupture is assumed to occur when the reactor is at power. The primary coolant boundary is reached. The initial leak rate of primary coolant through each end of the broken tube is assumed 30 lb/sec , and gradually decreases as the pressure difference between the reactor vessel and the SG is reduced. This leak rate is larger than the maximum capacity of the charge pumps to maintain inventory and of the pressurizer heaters to maintain pressure. Thus both pressure and level in the pressurizer would decrease. At about 15 minutes post accident, either the low pressurizer pressure trip or the low pressurizer level set point is reached. The resultant reactor and turbine trip immediately terminate power output to the grid. This disturbance to the grid is assumed to cause loss of off- site power to the plant. With loss of off-site power, reactor cool down is affected by the operation of automatic safety valves and manual atmospheric relief valves.

Diagnosis of the Accident:

- Annunciation of condenser high radiation alarm
- Steam generator feedwater/steam flow mismatch.
- Decreasing in pressurizer level and pressure ,
- Increasing level in affected SG can cause reactor trip.

Operator Actions

We assumed that the operator response will take about 30 minutes from the beginning of the rupture to diagnose the accident .

- isolation the affected SG
- termination its emergency FW
- closing its steam relief valves
- plant recovery

Radiological Consequences:

The SG is isolated when its emergency FW is terminated and its relief valves are close . Before reactor trip and SG venting , some iodine is removed from the condensate by the air- ejector and released to the environment. Radioactivity released via this route is small due to preferential retension of iodine in the condensate. At 15 minutes post-accident, reactor trip and loss of offsite power terminate availability of the condenser and steam is vented through the safety/relief valves . At this time , leakage through the broken tube is decreased through each end (e.g. is down to 23 lb/ sec) . Only the jet of coolant pointing upward at the steam outlet contributes directly to offsite closes. This jet of water is assumed atomized into droplets having diameter in the micron range and carried by steam through the safety relief valves to the environment. Since droplets,carrying iodine at same level as that of the primary coolant, would be captured by the surrounding water in th SG, by the steam separator dryer and by other internal hardwater.

Conclusions and Recommendations

1. The TLFW is the major contributor in the overpressurization in the RCS, which is the beginning for the initiation of leak and may be rupture of some tubes of the UTSG.
2. The inherent safety of the RCS plays an important part in mitigating the severe consequences of the fortune of the RPS .
3. Avoiding the occurrence of the TLFW by improving the FW performance and availability .
4. The design pressure of the RCS will not be exceeded by applying more safety systems.
5. Good choice of SG U-tubes material according to ASME - code.
6. Simplicity in the separation of the failed components or system .
7. Introduction of an expert reactor protection system (ERPS) with two levels of reactor scram (Diversity approach) to avoid the failure of the (RPS), and to give early

event diagnosis, and control it automatically.

RESULTS AND DISCUSSION

The input parameters of the Robinson PWR nuclear power station [8] were introduced in the program of the simulation model which was derived from the mathematical model. The disturbed parameter in the model is the termination of the main feedwater to all steam generators, and assuming the failure of the reactor protection system to scram the reactor at any trip signal. Assuming the interval of the transient was 500 second, the results of this event on the important parameters can be represented by Figures 1-4. Figure 1 shows the increase of the average reactor coolant temperature which reaches to maximum increase by amount of 80 °C above the steady state value. This increase in the reactor coolant is due to the reduction of the heat removal from the primary reactor coolant system to the secondary side of the steam generators.

Figure 2 illustrates the continous reduction of the core power due to the coolant temperature increase which results in negative moderator reactivity feedback .

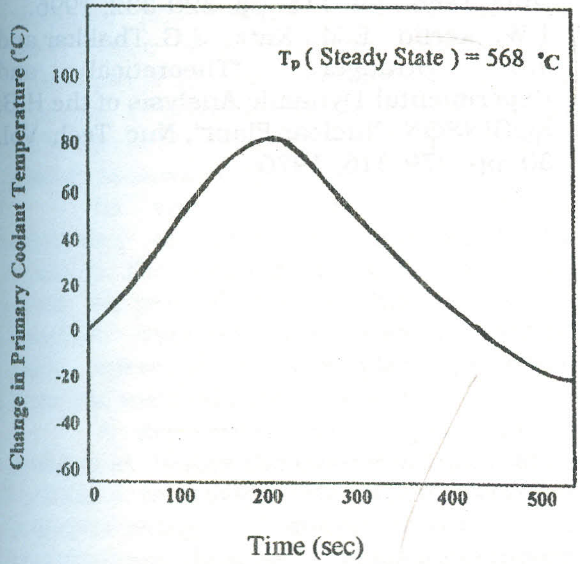


Figure 1 Change in primary coolant temperature during LOFW

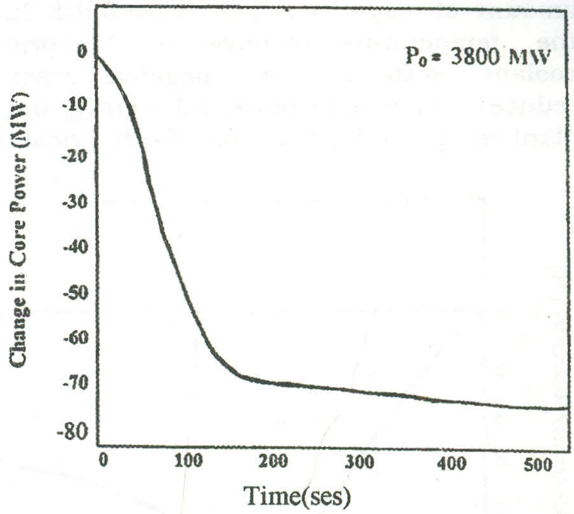


Figure 2 Change in core power during LOFW

Figure 3 illustrates the high pressure increase in the primary coolant system, due to the thermal expansion of the reactor coolant which causes an insurge of subcooled primary coolant into the pressurizer. The water and steam phases in the pressurizer exist because of the relatively large height to diameter ratio, but the two share remain in nonequilibrium state. Increased reactor coolant pressure produces activation of the pressurizer sprays, however, their flow rate do not terminate the pressure rise.

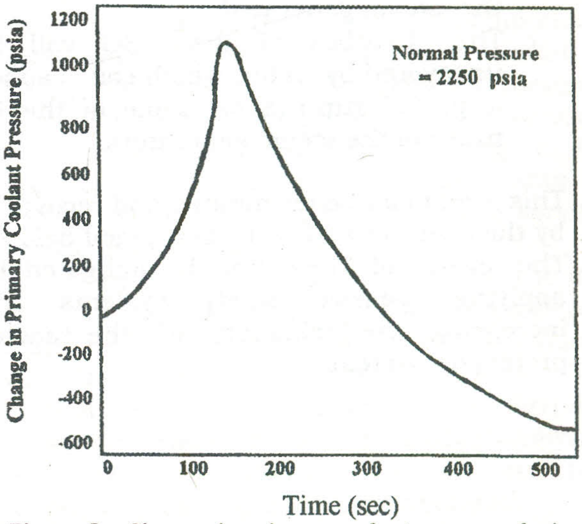


Figure 3 Change in primary coolant pressure during LOFW

Figure 4 represents the large change in the amount of negative reactivity feedback due to the temperature increase of the primary coolant system. This negative reactivity reduces the reactor power inherently, and the plant can go to the direction of subcriticality.

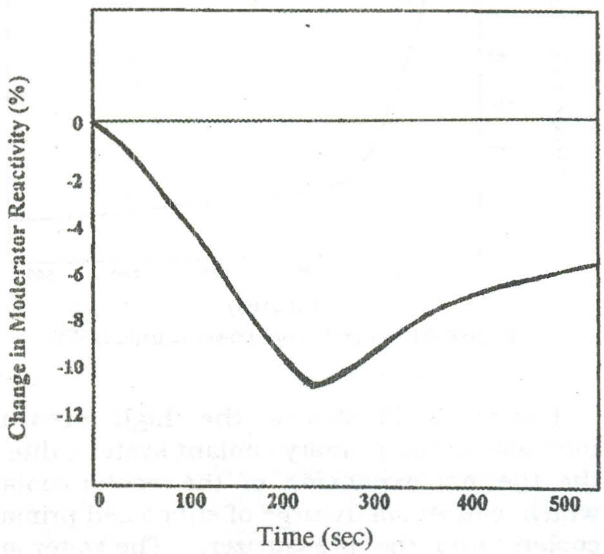


Figure 4 Change in Moderator Reactivity Feed back During LOFW

CONCLUSION

1. We conclude that the total loss of the main feedwater will result in:
   a) The large rise of temperature and pressure of the primary coolant system.
   b) The loss of heat sink in the shell side of the steam generated.
   c The U-tubes of the SG will be uncovered by water, which can cause a partial rupture for some of the U-tubes of the steam generators.

2. This event can be terminated and recovered by the operator action as mentioned before.
3. The event of TLFW can be mitigated by applying passive safety systems and increasing the reliability of the reactor protection system.

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