

TASKS:

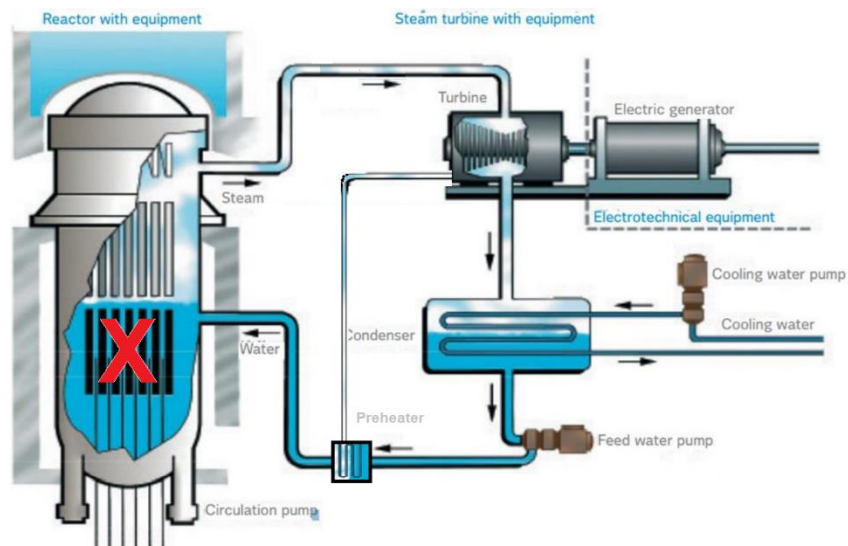
- 1- Consider the following two scenarios from your exercise cases,
- 2- Describe briefly how you do expect the system to behave during each transient. (how the parameters that you consider to be relevant such as power, pressure, flowrates, void, levels, temperatures, etc. will change?)
- 3- Send a report with your prediction and explanation for these cases to me.
- 4- As soon as I receive your report, I will send the password for the file "SH2705_2023_Alt-exam-plots" containing plots of different parameters to you.
- 5- Study the graphs of the event and compare the results with your prediction, is everything according to your prediction? If not explain what you think is the reason for these discrepancies.
- 6- Complete your report, prepare a presentation for us to have a zoom meeting for final questioning and grading.
- 7- You need to send your reports to me before July 27th and have your presentation before July 29th if you need your credits to be registered before July 31st.

The model

The reactor is a generic Nordic Boiling Water Reactor (BWR), with a core length of 3.7 meters and a nominal power at 2500 MW. The neutron flux is approximated with 2 energy groups (i.e. fast and thermal neutrons) using the diffusion approximation. The time dependent behavior of the reactor is referred as the reactor dynamics and consist of 6 delayed neutron feedbacks. Decay heat is simulated by the ANSI/ANS-5.1-1979 standard. This decay heat is correlated to the nominal power. The reactor simulation model is constructed with several scrambling mechanisms, some directly coupled from accident initiating events, and other by deviations of key process parameters. Values for different plant parameters and SCRAM limits are as follows

Parameter	Value
Power	2500 MWth
Pressure steam dome	70 MPa
Recirculation flow rate	7800 kg/s
Feedwater flow rate	1350 kg/s
Water level above the core	4.3 m
Feedwater temperature	185° C
Safety Release Valves flow rate	1200 kg/s
Property	SCRAM limit
High nominal power	106 %
Low coolant level	> 3.1 m
High coolant level	< 5.0 m
Steam Pressure	> 7.4 MPa
Wetwell liquid temperature	> 35° C

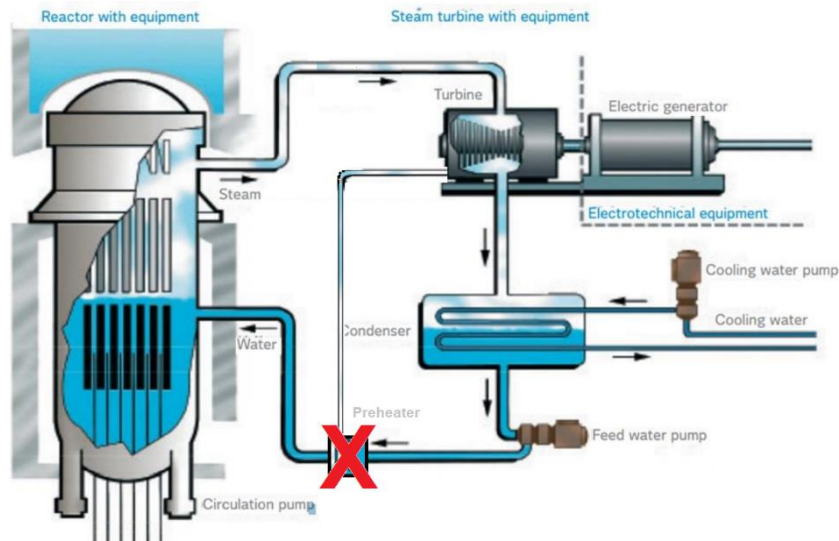
Case 1- Manual activation of the reactor SCRAM



Assume that the reactor is manually scrammed by operators during normal operation at nominal power.

Contemplate what will happen with the system? How will the power and pressure behave? How will the main recirculation flow, feed water flow, steam flow, void and water level in the vessel change? what will happen to the auxiliary feed water system and ADS system?

Case 3- Feed water enthalpy decrease



Consider loss of feedwater preheaters during normal operation at nominal power which will result in reduction of the temperature in the feedwater. We assume a very slow rate of temperature decrease from 185°C to 161.5°C. during 60 seconds to avoid scram. (consider the Oskarshamn 2 1999 event)

Contemplate what will happen with the system? How will the power and pressure behave? How will the main recirculation flow, feed water flow, steam flow, void and water level in the vessel change? what will happen to the auxiliary feed water system and ADS system? Will this effect the coolant temperature in the core? If yes to which extent?