LECTURE 9: SAFETY ANALYSIS REQUIREMENTS

MODULE OBJECTIVES:

At the end of this module, you will be able to describe the critical core conditions and safety system performance assumptions in a :

1. **Large loss of coolant accident analysis**
2. **In-core loss of coolant accident analysis**

- The role of physics analysis in large LOCA is to determine the power pulse due to the reactivity transient and the energy deposition in fuel. To maximize the effect of the potential power pulse consequences, certain assumptions are made of the initial core state and in the analysis methodology. These assumptions place the shutdown system performance under the most severe tests using a combination of worst but credible conditions.

Pre-Event Reactor Conditions

- The accident is assumed to occur at a time when the moderator is heavily poisoned, i.e. at the time of a restart after a prolonged outage and the adjuster banks are all withdrawn. The absence of the saturating fission products and the adjuster bank being withdrawn both require reactivity compensation by moderator poison.

- The pressure tubes are creeping diametrally and length-wise over their life time. The enlarged pressure tubes lead to higher coolant volume and higher void reactivity hold-up. Conservative estimates of the current pressure tube diameter increase due to creep are used in the lattice cell model and hence in the coolant void calculations.

- The reactor is assumed to be operated with coolant isotopic purity at its lower limit.

- A tilted flux shape existing at the time of the accident can aggravate the void effect The PHT configuration in CANDU 6 is such that a break in one pass will initially affect a quarter of the channels located to one side of the core. If a side-to-side flux tilt already exists and the high flux side coincides with the voiding side, then the void reactivity effect would be aggravated.

- Also, if a bottom-to-top flux tilt already exists at the time of accident, the effectiveness of the shutoff rods can be reduced since they take a longer time to reach the high-flux bottom of the core.

- Various initial tilted flux shapes are therefore assumed in LOCA analysis. Note also that the initial reactor power is reduced from

full power to avoid early reactor trip on high power signals from the in-core detectors.

Other Analysis Assumptions

Other assumptions that maximize the power pulse and its consequences are:

a. **Trip set-points are to include uncertainty allowance.**

b. **Trip time is to be based on the backup trip rather than the first trip, and on the third logic channel.**

c. **The two most effective shutoff rods (or one most effective LIZZ nozzle) are assumed non-operational. The two most effective rods are selected with respect to the break type and the location being analyzed.**

d. **Coolant void reactivity is deliberately augmented to allow for calculation uncertainty.**

e. Reactor Regulating System actions are ignored.

Break Types

- **Two break types are usually of the most interest - a large break (100% Pump Suction Break, or 100% Reactor Outlet Header Break) that leads to the highest energy pulse and energy deposition in fuel, and a critical break (about 20-30% Reactor Inlet Header Break) that leads to flow stagnation and most severe pressure tube temperature transient The standard definition of break size is twice the pipe cross-sectional area for a 100% break.**

Reactor Trip Time

- **The electronic circuitry for the neutronic trips are modelled in order to determine, as closely as possible, the actuation times of the shutdown systems. By comparing to trip set-points, the high-power trip time of each in-core detector and the rate-log-power trip time of each ion-chamber are determined. Trip of all three logic channels is demanded, i.e. at least one detector in each logic channel has tripped.**

- **Since the backup trip is to be credited, the shutdown system actuation time is then the later of the high-power and rate-log trip times.**

- **An instrument-loop uncertainty is also normally assigned to the rate-log trip set-points.**

Shutdown-System Effectiveness

- **The primary measure of shutdown-system effectiveness is the margin to fuel-breakup. The energy stored in the fuel is the sum of the initial stored energy (i.e. steady state energy content) plus the energy added by the power pulse.**

- **The highest allowable bundle power in CANDU 6 is 935 kW. To assess fuel integrity, the energy stored in the hottest fuel element of a 935 kW bundle is evaluated. This fuel element is assumed to be subject to the power pulse of an actual bundle with the largest time-integrated power up to 5 seconds.**

- **The total energy stored is then compared to a conservative lower limit required for fuel breakup, typically taken as 840 J/g of fuel.**

In-Core Loss-of-Coolant Accident Analysis

General System Behaviour

- **The postulated spontaneous rupture of a pressure tube while the reactor is operating at power, is one of the events assessed in order to evaluate the effectiveness of the special safety systems.**

- **The calandria tube surrounding the ruptured pressure tube is assumed to have also failed. Primary circuit coolant discharges into the calandria.**

- **The discharging hot coolant, and possibly with ejected fuel bundles, can cause structural damage, disabling some shutoff rod guide-tubes and MCA guide-tubes.**

- **The pressure and inventory control system cannot make up for the discharge and the primary circuit would depressuize; voiding would occur in all channels.**

- The discharging coolant will also mix with the moderator and dilute the poison concentration, and downgrade the moderator isotopic purity.

- The rate of voiding would be slow so that the reactor regulating system could compensate for the void reactivity. It is expected that low heat transport pressure trip and pressurizer low level trip set-points will be reached in 2-3 minutes.

- Moderator temperature increase would be relatively slow because of the high thermal capacity of the moderator.

Physics Considerations

- The coolant void reactivity insertion rate from the rupture of a channel is much smaller than that as in the case of a large LOCA.

- It has been often assumed that in a small break, up to the time of reactor trip, the regulating system will compensate for the void reactivity insertion, and maintain the reactor bulk power at the demanded level.

- The RRS may also drive the mechanical control absorbers in the core. In such cases the power distribution will be more distorted.

- Physics calculations in in-core LOCA accident analysis provide an evaluation of the shutdown system effectiveness, particularly in terms of sufficient depth of SDSI when the system is partially impaired.

- Furthermore, the transient reactor regulating system response before reactor trip and hence the power distribution distortion and variations with time can also be modelled in physics kinetics calculations.

SDS1 Depth

- After reactor trip, the shutoff rods are inserted and the reactor is sub-critical.

- The available number of shutoff rods may not be the full complement of the system - some rods are assumed not able to insert due to damaged guide tubes,

- The discharging coolant maintains the positive reactivity insertion after reactor shutdown. The shutdown system must be able to keep the reactor in a sub-critical state up to a time when operator intervention can be credited, which is accepted at after 15 minutes of an unambiguous alarm indication of the accident event

SDS 1 Depth Analysis Method

- The current analysis methodology is to simulate the reactor core at 15 minutes after the initiation of the accident, modelling the core configuration as closely as possible:

a. The moderator poison concentration as predicted by the most credible coolant discharge calculations and mixing model.

b. The moderator temperature as predicted by moderator pressure and temperature transient calculations.

C. The coolant density distribution in the four passes as predicted by thermalhydraulics transient calculations.

d. The insertion of available shutoff rods which are not damaged by discharging fuel and coolant as predicted by the most credible damage assessment.

e. The degradation of moderator isotopic purity due to mixing with the discharging coolant as predicted by the most credible mixing model.

- Mitigating actions from the reactor regulating system and other safety systems are often not credited in the analysis: emergency coolant injection and boiler crash cool-down are not credited, RRS action is ignored and not modelled.

- A highly poisoned moderator obviously leads to a more severe reactivity transient due to poison dilution. Also a highly poison moderator enhances the coolant void reactivity.

- Therefore the accident is postulated to occur at restart after a prolonged shutdown when the absence of fission product reactivity load is compensated by moderator poison. and the adjusters are out of core.

- Damage to the shutdown system is assessed according to the cause of the in-core breaks.

- The selection of the broken channel and the location of the break are chosen to maximize the damage in terms of the number of shutoff rods disabled and the relative effectiveness of these disabled rods.

- Coolant void reactivity is aggravated by low coolant isotopic purity. Therefore the lower limit on operating purity is assumed in the simulation to maximize the void effect

Mixing Models

- The dilution of poison in the moderator is calculated according to certain mixing models.

- With the "Piston Mixing" model, the discharging coolant is assumed to act as a "piston", displacing unmixed poisoned moderator which is expelled through the rupture discs.

- In the "Uniform Mixing" model, the discharging coolant is assumed to mix uniformly and instantly with the poisoned moderator so that the poison concentration of the expelled moderator is the same as the average poison concentration throughout the moderator

- The "Delayed Mixing" model is that the poison concentration of the fluid discharged through the rupture discs is equal to the average poison concentration at an earlier time T, which is the characteristic time over which the mixing takes place.

1. An "initial" poison concentration $[Gd]_0$ at the start of the accident is calculated. This corresponds to the poison needed to compensate for the excess reactivity of the zero-power, hot restart core state after a long shutdown, with all adjusters out.

2. A dilution factor DF_{15} corresponding to coolant discharge up to 15 minutes is applied. The nominal " diluted" poison concentration at the 15th minute is $[Gd]_d = [Gd]_0 / DF15$-

3. The "Critical" poison concentration $[Gd]_{15}$ for the core state at the 15th minute with PHT partially voided, moderator poison diluted, moderator temperature increased and partial set of SOR inserted, is calculated. The margin to criticality is therefore given by

$$M = [Gd]_0 / DF_{15} - [Gd]_{15}$$

- An assessment must be made to determine any bias error and random uncertainty in M. Any bias error so determined should be applied to adjust the margin. The margin to criticality is then measured in units of sigma, which is one standard deviation of the random uncertainty.

- Generally speaking, the uncertainty in these calculated quantities can be estimated through comparisons to corresponding measurement data.

- The uncertainty estimate for $[Gd]_{15}$ is based on the reactivity components introduced by the perturbations as the abnormal core conditions, and accuracy of RFSP capturing the reactivity effects of these perturbations.

Sample Results of SDS1 Depth Analysis

- For illustration purposes, the results from a recent study for CANDU 6 plants for the case of a pressure-tube and calandria-tube rupture event are described below.

- The initial core at time zero corresponded to a restarted core after a prolonged shutdown at zero power hot conditions. All adjusters were withdrawn. To further increase the excess reactivity, fuelling ahead of 5 mk while the reactor was shut down was assumed.

The critical poison level $[Gd]_0$ was calculated to be 6.86 ppm boron.

- **Channel E11 was assumed to rupture at time zero. The PHT blow-down was computed by SOPHT, and the thermalhydraulics conditions at 907 second were modelled in RFSP "all-effects-included" simulation.**

- **A total of six SOR's were assumed non-operational: five disabled rods and one additional unavailable rod.**

- **The moderator D_20 isotopic purity was 99.94 atom percent, and was not degraded by the coolant discharged. However, in the coolant void effect calculation, the coolant isotopic purity was degraded to 95.08 atom percent to enhance the void reactivity.**

- **The just critical poison level $[Gd]_{15}$ at 907 s was determined to be 3.16 ppm boron.**

- **The delayed mixing model was used to compute the dilution factor with a characteristic time of 15 s. The dilution factor DF_{15} was determined to be 1.38. Therefore the diluted poison level $[Gd]_d$ at 907 s would be 6.86 / 1.38 = 4.98 ppm boron.**

- **Comparing this to the just critical poison level of 3.16 ppm boron, there is a safety margin of 1.82 ppm boron, which is equivalent to about 15 mk.**

- **The adequacy of such a margin was judged in the context of calculation uncertainty. The 1-σ uncertainty in $[Gd]_d$ was estimated to be ± 14%. The uncertainty in $[Gd]_{15}$ was estimated to be also ± 14%. The 1-σ uncertainty in the safety margin was then given by $[(4.98 \times 0.14)^2 + (3.16 \times 0.14)^2]^{1/2} = \pm 0.83$ ppm boron. The safety margin is more than two-sigma and hence there is greater 98% probability that the reactor remains sub-critical.**

Guaranteed Shutdown State Poison Requirement

- Closely related to in-core break physics analysis is the determination of the guaranteed shutdown state (GSS) poison requirement When the reactor is shut shown, from safety viewpoint, the moderator poison level should be such that the reactor is guaranteed to be sub-critical under all conditions.

- The safety concerns are addressed by postulating a most reactive core state with a combination of abnormal accident conditions - an in-core break leading to poison dilution, moderator temperature increase, and complete PHT voiding.

- The SDS1 depth analysis and the GSS poison requirement analysis share a lot of common elements - an in-core LOCA diluting the poison and coolant voiding as the most limiting scenario. However, there are some essential differences: the SOR's are not inserted in GSS, and there is no 15-minute time frame. Thus the PHT is assumed to be completely voided, and the dilution is with all available PHT inventory.

Sample results

- The following values are typical of a CANDU 6 plant and are used for illustrative purposes only. The critical moderator poison concentration was determined by RFSP simulation of the "most reactive" core to be 10.5 ppm boron. The nominal GSS poison requirement is therefore 27.3 ppm boron.

- Preliminary uncertainty assessments obtained a Uncertain Factor UF 1.38. The GSS poison requirement was therefore 27.3 x 1.38 = 38 ppm boron.

Compliance to Licensing Power Limits

- The current practices in CANDU 6 plants are to demonstrate compliance through simulations of reactor operations which are carried out at frequent intervals using reactor physics codes and models.

- It is recognized that these simulations have inherent errors and it is important that the magnitude of these errors is carefully determined and justified, and factored in the comparisons to the licensing limits.

- To allow for these uncertainties, certain administrative power limits are defined in the operating procedures to assist in ensuring compliance. The fuelling engineer makes every effort to ensure the fuelling schedule results in peak powers below these administrative limits.

- The operating history versus performance targets in terms of transgressions above these administrative limits are carefully tracked and analyzed. The frequency at which the compliance calculations are carried out is currently 2 or 3 times a week in a CANDU 6 plant.

- The compliance calculations are performed usually at a time with xenon at equilibrium with flux distribution, i.e. at a time when transient xenon effects in the refuelled channels have settled. Thus the calculated maximum channel power and bundle power used for compliance should have allowance not only for uncertainties in the calculation, but also for transient powers between surveillance times.

- As an example to provide more details on the compliance analysis and procedure, the current simulation method, error allowances, transient power variation estimates, and refinement in methodology being developed for Point Lepreau are discussed in the following subsections.

Simulation Method

- The flux / power mapping option in RFSP is used for core tracking purposes at Point Lepreau. It is based on best fitting the 102 in-core vanadium detector readings by a linear combination of a set of pre-calculated basis functions which are eigen-functions of the two-group diffusion equation. The fundamental flux shape function corresponds to a solution obtained for the latest core configuration.

- The calculations are done every Monday and Thursday morning. Typically refuelling of 7-10 channels starts after each calculation, and the xenon transient effects would have settled by the time of the next calculation. The sum of the mapped bundle powers is normalized to total reactor power.

Steady State Calculation Errors

- There are inherent errors in the flux and power mapping process: detector measurement errors, detector position uncertainty, accuracy and completeness of the basis shape functions, uncertainties introduced by detector flux interpolation from the mesh fluxes, and conversion of cell fluxes to bundle powers.

- Furthermore, there are limitations of the mapping calculation as well. The in-core detectors are located in the central core region and do not cover the peripheral region; the harmonic flux shapes in the flux synthesis have inherent errors due to core modelling approximations and the diffusion method.

- Normalization to the total reactor power also introduces errors since there are uncertainties in the measured total power.

- The assessment of the mapped flux error is based on comparisons to special flux scan measurements using a travelling miniature fission chamber.

- The assessment of the mapped channel power error is based on comparisons to heat balance data derived from predicted coolant flow rate and measured temperature increase.

- The measurement data itself has uncertainty and must be considered as well AR known sources of errors are identified and examined, and their contribution to the net error quantified.

- Furthermore, possible correlations between the various error terms to core physics parameters (such as fuel burnup) or to core model uncertainty (such as adjuster position) are investigated.

- An interim channel power and bundle power calculation uncertainty of ± 2.7% has been in use at Point Lepreau for compliance analysis purposes.

- It represents the channel and bundle calculation random one-sigma uncertainty. The administrative limits are set at one-sigma and two-sigma level below the licensing limits.

Transient Powers

- Transient power distributions due to xenon-free effects are estimated by means of corrections to the steady powers. These corrections are applied to the refuelled channels and their immediate eight neighbors.
- The correction factors were derived from detailed simulation studies of power transients after refuelled and comparing the power just after refuelling to the equilibrium power. The magnitude is of the order of a few percent, and is dependent on the location of the refuelled channel in the core.
- After the applications of the xenon-free corrections simultaneously to all affected channels, a transient power map representing the highest possible powers for each channel in between the surveillance times is created. Compliance statistics for this transient power map are also compiled.

Conclusion

- **Physics analysis has an essential role in defining the shutdown system performance requirements in terms of both speed and depth, and to demonstrate that the SDS's as designed can effectively mitigate any reactivity excursion or reactivity increment in credible accident scenarios.**

- **The material presented also illustrates the general approach in physics analysis - the philosophy of defining a worst possible core state leading to the most severe consequences and most stringent demands on system performance, and making the most unfavorable assumptions in the analysis process.**

- **In transient accident analysis, it is the neutronic kinetics behavior that drives the power variations. The neutron kinetics is directly affected by the changing core conditions, reactivity feedbacks and device movements. Thus physics analysis is closely feed to changing thermalhydraulic conditions and regulating system responses.**

- **Physics analysis also plays an indispensable role in meeting routine operation requirements. Physics simulations of the reactor core operation give essential performance data such as channel and bundle power distributions which are used to ensure compliance to licensing power limits.**

- **It has also been shown that the uncertainties in the calculation results are important elements in the assessment of safety margins and in providing a high level of confidence of the analysis conclusions.**