HOL90

APPENDIX 8

Lecture 8.1

CANDU TRAINING COURSE ON

NSSS DESIGN AND ANALYSIS

ACCIDENT ANALYSIS OVERVIEW

by

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1. INTRODUCTION

Historically, deterministic accident analyses have been used in the licensing of CANDU nuclear power plants in Canada. This has followed the single/dual failure approach:

1. a failure in a process system with all safety systems available, and
2. a failure in a process system with co-incident impairment in one of the safety systems.

Plant safety is assured by demonstrating that the consequences of a broad spectrum of postulated events satisfy acceptance criteria derived from regulatory requirements. These events fit into Category A as described in the CANDU 6 Licensing Basis Document (LBD), i.e. events used in the design and performance assessment of safety systems.

The LBD also refers to Category B events which are analyzed by probabilistic methods. This is discussed further in subsequent lectures. At present, it is sufficient to say that event tree/fault tree methods are used to determine frequencies of event sequences. Supporting accident analyses are done to determine the consequences of event sequences.

2. THE PURPOSE OF ACCIDENT ANALYSIS

As mentioned in the introduction there are two main reasons for doing Category A deterministic accident analyses:

1. to assist in the design of safety systems, and
2. to assess the performance of the safety systems once designed.

During the conceptual design phase, accident analyses are performed to see if the conceptual design can meet safety requirements, or whether design modifications are needed. Key design parameters of the safety systems and the event which sets the requirement for each parameter are given in Table 1. The shutdown systems trip parameter setpoints are determined by the events shown in Table 2. For example, the large loss-of-coolant accident (LOCA) sets the speed requirement for the shutdown systems, while, the pressure tube rupture and consequential failure of its calandria tube when the moderator is highly poisoned sets the shutdown systems depth requirement. When the detailed design of systems is complete, a check on the performance of the safety systems is made by a final licensing analysis.

Category B accident analyses are done as a part of the overall Probabilistic Safety Assessment to show that event sequences meet frequency/consequence (dose) criteria.

3. ANALYSIS METHODS

Deterministic accident analysis methods have evolved over the years. Sophisticated computer codes which have been developed and validated over a period of several years are used. Pessimistic assumptions are used in Category A analysis so that a conservative assessment of safety system performance is made.

The analysis process starts by defining the assumptions, methodology and acceptance criteria to be used in a Safety Analysis Basis (SAB) document. A sample SAB from past analysis is attached as Appendix A. Assumptions are given for the process systems, the safety systems, and the general analysis methodology. As an example, some of the pessimistic assumptions made for Category A analysis are:

1. The two most effective rods of shutdown system No. 1 are assumed unavailable.
2. The most effective poison injection nozzle of shutdown system No. 2 is assumed unavailable.
3. The least effective trip on the least effective shutdown system only is credited.
4. The reactor regulating system action is not credited if it improves the event sequence (i.e., no credit for set back or step back action).
5. 103 percent full power to account for possible power measurement error is used.
6. Operator action before 15 minutes is not credited.
7. Instantaneous guillotine cross-sectional rupture for discharge rate calculations is assumed.
8. Pasquil F weather conditions are assumed to exist during the first hour of radioactive releases.

For Category B analysis, the system assumptions are determined by the specific event sequence under study. Generally though, the assumptions reflect a more realistic assessment of plant response to the initiating event.

The analysis methodology section of the SAB describes what analysis will be done and what computer codes will be used. The overall accident analysis involves some or all of the following analysis disciplines:

1. reactor physics analysis determines the reactor, fuel channel, fuel bundle, and fuel element power transients and neutronic trip times; also determines fuel element power versus burnup used in the fuel analysis,
2. thermalhydraulic analysis determines transient behaviour of the heat transport system such as flows, pressures, void fractions and heat transfer conditions; used as input to reactor physics, fuel, fuel channel, and containment analyses,
3. fuel analysis determines the transient fuel and fuel sheath temperatures and whether the fuel sheath fails and the fission product release from failed fuel,
4. fuel channel analysis determines the pressure tube and calandria tube behaviour and whether the fuel channel fails,
5. containment analysis determines the transient temperatures and pressures in containment, and fission product releases from containment,
6. atmospheric dispersion analysis determines the spatial and temporal fission product concentrations outside containment for use in the dose calculations, and
7. dose analysis determines individual and population radiation doses.

Acceptance criteria are specified against which the analysis results are compared. These are Atomic Energy Control Board (AECB) criteria or derived from these. For example, AECB dose limits are given in Table 3 for Category A events. Additionally, the AECB require no fuel failures be predicted for single failures, other than large loss-of-coolant accidents, and that fuel channel integrity be maintained for all events. The safety analyst may use more stringent criteria, such as no fuel sheath dryout instead of no fuel sheath failure or no calandria tube dryout on pressure tube contact instead of no fuel channel failure. These more stringent, sufficient criteria may be used to provide a degree of conservatism or to avoid more detailed, costly and potentially uncertain analysis.

4. ANALYSIS OVERVIEW

This section gives an overview of the events to be analyzed in Category A for the performance assessment of the safety systems. Consequence analysis to be performed in support of the Probabilistic Safety Assessment (Category B) will be determined in the course of doing the probabilistic studies.

The two major performance checks of the safety systems are:

1. to ensure that the integrated performance of the safety systems, in response to an event during operation, is such that dose limits and other criteria are not exceeded, and
2. to ensure that adequate shutdown systems trip coverage is provided under a wide range of potential operating conditions.

Events which test the integrated performance of shutdown, emergency core cooling and containment systems are listed and discussed in subcategory A.1. The set of events which are used to demonstrate adequate trip coverage is listed and discussed in subcategory A.2.

Some analyses are performed which go beyond the objectives of subcategories A.1 and A.2. These special analyses are performed to evaluate margins in the design, and are discussed in subcategory A.3.

The following families of events are considered to help select events which might establish design requirement for safety systems:

1. loss of heat transport system coolant inventory,
2. loss of heat transport system coolant flow,
3. loss of reactor control functions,
4. loss of reactor heat sink, and
5. moderator and end shield failures.

The following events are obtained from the loss of heat transport system inventory family:

1. in-core breaks (pressure tube rupture), and
2. out-of-core breaks (end fitting failure, small and large losses of coolant).

The loss of heat transport system coolant flow yields:

1. total loss of Class IV power (loss of all pumps),
2. partial loss of Class IV power (loss of one pump),
3. pump seizure, and
4. single channel flow reduction.

Loss of reactor control functions gives rise to:

1. loss of reactivity control, and
2. loss of pressure or inventory control.

Loss of heat sink gives two events:

1. steam line breaks, and
2. feed line breaks.

Moderator and end shield failures include:

1. loss of service water to the moderator heat exchangers,
2. pipe break in the moderator system, and
3. loss of end shield cooling.

4.1 SUBCATEGORY A.1 – SAFETY SYSTEM PERFORMANCE

The loss of heat transport coolant inventory requires action from all safety systems. Events from this family make up most of the subcategory A.1 events. A single channel flow reduction is included because severe flow reduction can lead to fuel channel failure, hence a loss of heat transport coolant inventory. Loss of reactivity control is included because it sets the requirements for the in-core neutronic detector locations for the shutdown systems. A steam line break inside containment is included because it sets a requirement for containment structural integrity.

Table 4 lists the events which must be evaluated to assess the performance of the safety systems. It lists not only the single process failures, but also the same failures coincident with a safety system impairment. Only those events marked 'X' will be evaluated. Events marked 'NR' are not relevant or are trivial.

4.2 SUBCATEGORY A.2 – SHUTDOWN SYSTEMS TRIP COVERAGE

The events evaluated for trip coverage are given in Table 5. The general requirement is to provide two effective trip parameters on both shutdown systems, where practicable. Criteria defining trip effectiveness are derived from AECB requirements. These criteria depend on the event being analyzed, and will be specified in the Safety Analysis Basis documents. Some examples are:

1. prevention of fuel sheath failures for a small LOCA,

2. prevention of heat transport system overpressurization for a loss of Class IV electrical power, and
3. prevention of channel failure due to overheating for a large LOCA.

The actual list of analyses performed to demonstrate adequate trip coverage is much larger than given in Table 5 due to the finer subdivision of events. For example, a feedline break may be considered upstream or downstream of the check valves. Any further subdivision will be detailed in the safety analysis basis documents.

4.3 SUBCATEGORY A.3

Subcategory A.3 events are those events which do not fit the precise definition of subcategories A.1 or A.2, but have been traditionally analyzed using the conservative tools and assumptions of Category A to determine margins in the safety systems designs. Typically, these are events for which the combinations of failures places them in a category with a lower frequency of occurrence than for the events considered previously.

4.3.1 Loss of Coolant and Loss of Class IV Electrical Power

The events to be considered under this subcategory are given in Table 6. They include small and large LOCA, with and without safety system impairments. The postulated event sequence is a loss of coolant, followed by a reactor trip and consequent turbine trip. Following this, electrical failures can occur which prevent the plant from receiving power from an outside source. The frequency of occurrence for this combination of events will be calculated in an appropriate probabilistic safety assessment study.

4.3.2 Special Containment Impairments

In the past, events listed in subcategory A.1 have also been analyzed with the following special containment impairments to quantify the margin available in the containment design:

1. open airlock doors,
2. total loss of dousing.

The airlocks during normal operation are closed and are permitted to be open only under restricted circumstances, with precautions taken. Hence, the probability of a loss of coolant coincident with both airlock doors open is remote.

The dousing system has two subsystems. The subcategory A.1 impairment is the failure of one subsystem. In subcategory A.3, we consider the more remote failure of both subsystems simultaneously.

The matrix of events to be evaluated with the above special containment impairments is given in Table 7.

TABLE 1

SAFETY SYSTEM DESIGN PARAMETERS

AND EVENTS WHICH IMPOSE REQUIREMENTS

DESIGN PARAMETER	EVENT
SHUTDOWN	
Speed	Large loss of coolant
Depth	Pressure tube rupture
Detector location	Loss of reactivity control
ECC	
Initiation/Conditioning	Small loss of coolant
Initiation pressure	Large loss of coolant
Injection pressure	Small loss of coolant
Accumulator volume	Large loss of coolant
Injection flowrate	Large loss of coolant
CONTAINMENT	
Isolation setpoint pressure	Small loss of coolant
Isolation response time	End fitting failure
Design pressure	Large loss of coolant
Containment structural integrity	Steamline failure
Leakage rate	End fitting failure and loss of coolant coincident with loss of emergency core cooling.

TABLE 2

SHUTDOWN SYSTEM TRIP PARAMETERS AND

EVENTS WHICH IMPOSE REQUIREMENTS

TRIP PARAMETER	EVENT	
	SHUTDOWN SYSTEM NO. 1	SHUTDOWN SYSTEM NO. 2
Low flow	Single pump trip	N/A
Low core ΔP	N/A	Loss of grid power
High heat transport system pressure	Loss of grid power	Loss of grid power
Low heat transport system pressure	Small loss of coolant	Small loss of coolant
High reactor building pressure	Small loss of coolant	Small loss of coolant
Boiler low level	Feedline break	Feedline break
Boiler feedline low pressure	Steam main failure	Steam main failure
High moderator temperature	Loss of moderator cooling	N/A
High neutron power	Slow loss of power control	Slow loss of power control
Log rate	Large loss of coolant	Large loss of coolant
Low pressurizer level	Small loss of coolant	Small loss of coolant

TABLE 3

REFERENCE DOSE LIMITS FOR SINGLE/DUAL FAILURE EVENTS

EVENT	INDIVIDUAL DOSE LIMIT	POPULATION DOSE LIMIT
Single Failure	5 mSv whole body	100 man-sieverts, whole body
	30 mSv thyroid	100 man-sieverts, thyroid
Dual Failure	250 mSv whole body	10^4 man-sieverts, whole body
	2500 mSv thyroid	10^4 man-sieverts, thyroid

TABLE 4

SUBCATEGORY A.1 EVENTS (SINGLE/DUAL FAILURE ANALYSIS)

	SDS IMPAIRMENTS		ECC IMPAIRMENTS		CONTAINMENT IMPAIRMENTS[1]			
EVENT	SDS1	SDS2	INJECTION AND CRASH COOL	LOOP ISOLATION	ISOLATION DAMPERS	DOUSING	LOCAL AIR COOLING	DEFLATED AIRLOCK SEALS
Large loss of coolant	X	X	X	X	X	X	X	X
Small loss of coolant	X	X	X	X	X	X	X	X
Steamline break	X	X	NR[2]	NR[2]	NR[3]	X	X	NR[3]
Pressure tube rupture	X	X	X	X	X	X	X	X
End fitting failure	X	X	X	X	X	X	X	X
Loss of reactivity control	X	X	NR[4]	NR[4]	NR[4]	NR[4]	NR[4]	NR[4]
Single channel flow reduction	X	X	X	X	X	X	X	X
Steam generator tube rupture	X	X	X	X	NR[5]	NR[5]	NR[5]	NR[5]

NR = not relevant, see notes for explanation

X = analysis to be performed

Notes for Table 4:

3. Containment impairments are as follows:
   a. isolation dampers - failure of isolation logic (all dampers in inlet and outlet ventilation system ducting fail to close),
   b. dousing - failure of one subsystem,
   c. local air cooling - failure of all local air coolers inside the reactor building, and
   d. deflated airlock seals – failure of seals in both inner and outer doors of airlock.
4. ECCS is not initiated for a steam main break outside containment and, therefore, its impairments are not relevant.

   ECCS is initiated and beneficial for steam main break inside containment. However, it is not credited in the analysis and, therefore, its impairments are not relevant.
5. It is conservative to assume isolation of containment for a steam main break inside containment. Therefore, its impairment is not considered.

   Containment impairments are not relevant for the steam main break outside containment because containment systems are neither required nor initiated.
6. ECCS or containment are not initiated or credited. Therefore, these impairments need not be considered.
7. This event leads to loss of coolant outside containment; therefore, containment impairment is not relevant.

TABLE 5

SUBCATEGORY A.2 EVENTS (TRIP COVERAGE)

1. Loss of Class IV power
2. Single heat transport pump trip
3. Single heat transport pump seizure
4. Small loss of coolant
5. Pressure tube rupture
6. Large loss of coolant
7. Loss of reactivity control
8. Loss of primary circuit inventory control
9. Loss of primary circuit pressure control
10. Loss of secondary circuit pressure control
11. Feedwater line break
12. Steam line break
13. Loss of service water to moderator
14. Moderator system pipe break
15. Loss of end-shield cooling

TABLE 6

SUBCATEGORY A.3 EVENTS - LOCA WITH LOSS OF CLASS IV POWER

	SDS IMPAIRMENTS		ECC IMPAIRMENTS		CONTAINMENT IMPAIRMENTS		
EVENT	SDS1	SDS2	INJECTION AND CRASH COOL	LOOP ISOLATION	ISOLATION DAMPERS	DOUSING	LOCAL AIR COOLING
Large LOCA	X	X	X	X	X	X	X
Small LOCA	X	X	X	X	X	X	X

Note: Containment impairments are as follows:

1. Isolation dampers - failure of isolation logic (all dampers in inlet and outlet ventilation system ducting fail to close),
2. dousing - failure of one subsystem, and
3. local air cooling - failure of all local air coolers inside the reactor building.

TABLE 7

SUBCATEGORY A.3 EVENTS (SPECIAL CONTAINMENT IMPAIRMENTS)

	TOTAL LOSS OF DOUSING	OPEN AIRLOCK DOORS
Large LOCA	X	X
End fitting failure	X	X