IAEA Safety Assessment Education and Training (SAET) Programme

Joint ICTP-IAEA Essential Knowledge Workshop on Deterministic Safety Assessment and Engineering Aspects Important to Safety

Postulated Initiating Events Exercise

IAEA
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Exercises on Postulated Initiating Events (PIEs)

Relevant information about PIEs and their categorization to be used in exercise can be found in IAEA SRS-30, pages 4 to 7. Download the IAEA SRS-30 from:

http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1162_web.pdf

Goal of the exercise is to get familiar with the process of the identification of the postulated initiating events, evaluation of their completeness and confirmation of their proper categorization. Below, excerpts for 3 different types of reactors are provided from their safety related documentation. Few changes were intentionally introduces or some parts of the documentation were left out for the purposes of the exercise.

Tasks to be performed

1. Download IAEA Safety Report Series no. 30 Accident Analysis for Nuclear Power Plants with Pressurized Water Reactors, and use this document as a basis for your review.
2. As regulator facing review and approval of a safety case documentation provided by a vendor read carefully the provided examples of parts of Chapter 15 safety analysis reports related to PIEs.
3. Outline the review procedure and define the main aspects of your review.
4. Review the provided information on PIEs and their categorization. Focus on:
    a. Completeness of PIEs
    b. Basis for their categorization and indicated categories of PIEs
5. Identify your next actions as regulator – will you accept the presented PIEs for the safety analyses or will you have additional questions to the vendor or operator.
6. Prepare a presentation summarizing and discussing your findings for each example.

Example 1

Type of reactor: PWR

Categorization used in reviewed safety related documentation:

- Category 1 – steady-state and normal operation conditions
- Category 2 – anticipated operational occurrences with a frequency of occurrence more than 10^{-2} events per year (in the most severe case they can lead to reactor shutdown after which the operation of the station can be resumed. States of such a kind have no tendency for extension to include the risk of more severe failures, i.e. the design basis conditions of category 3 or 4);
- Category 3 – accident conditions with frequency of occurrence ranging from 10^{-2} to 10^{-4} events per year (in these conditions a damage of not more than 1 % of total quantity of fuel rods is only possible);
- Category 4 – accident conditions with frequency of occurrence ranging from 10^{-4} to 10^{-6} events per year (these conditions are the most severe cases out of design conditions. Protective measures are to be envisaged in the design against them. The damage of not more than 10 % of the total number of fuel rods is only possible).

List of PIEs:

Following groups of PIE are considered:

- increase in heat removal through the secondary circuit;
- decrease in heat removal through the secondary circuit;
- decrease in the primary coolant flowrate;
- reactivity and power distribution abnormalities;
- increase in the primary coolant inventory;
- decrease in the primary coolant inventory;
- leak into the environment or the secondary circuit system;
- fuel mishandling;
- radioactive releases from the systems and equipment of other circuits and systems;
- spurious actuation of systems

Categorization of PIEs:

Following PIEs are categorized as category 2 events:

Increase in heat removal through the secondary side
15.1.1 Feedwater system malfunction resulting in feedwater temperature decrease
15.1.2 Feedwater system malfunction resulting in feedwater flow increase
15.1.3 Inadvertent opening of steam generator safety valve, dump valve (BRU-A) or turbine by-pass valve (BRU-K) followed by their failure to seat
15.1.4 Increase in steam flowrate to the turbine (due to malfunction or failure of the steam pressure controller) (instant increase in the turbine load by 10 % above the nominal)

Decrease in heat removal through the secondary circuit
15.2.1 Decrease in steam flowrate to the turbine (due to malfunction or failure of the steam pressure controller)
15.2.2 Closing of the turbine stop valves or loss of external electrical load
15.2.3 Loss of normal feedwater flowrate (with the exception of the feedwater pipeline break)
15.2.4 Spurious closing of MSIV
15.2.5 Loss of non-emergency a.c. power to the station auxiliaries (loss of NPP power for 2 and 72 hours)

Decrease in primary coolant flowrate
15.3.1 Trip of different number of reactor coolant pump sets
15.3.2 Emergency deviation of the grid frequency

Reactivity and power distribution abnormalities
15.4.1 Uncontrolled withdrawal of control rod group at HZP and at power
15.4.3 Operator's error in xenon oscillation suppression (moving of control rods of the control and protection system that causes the greatest possible power tilt)
15.4.5 Inadvertent dilution of boric acid in the primary coolant
15.4.6 Mismatching of CPS ARs:
- drop of CPS AR;
- drop of CPS AR group or subgroup;
- static mismatching in CPS AR group

Increase in the primary coolant inventory
15.5.1 Malfunction in the chemical and volume control system leading to increase in the primary coolant inventory caused by water injection
15.5.2 Inadvertent actuation of quick boron injection system
15.5.3 Spurious water injection into the pressurizer

Decrease in the primary coolant inventory
15.6.4 Small leaks from the primary circuit compensated by the normal make-up system, including rupture of instrumentation line

Fuel mishandling
15.8.2 Damage of spent fuel pool cooling system

Spurious operation of systems
15.10.1 Spurious scram
15.10.2 Inadvertent actuation of the emergency core cooling system
15.10.3 Inadvertent actuation of the passive heat removal system
15.10.4 Inadvertent actuation of the steam generator blowdown and emergency cooldown system
15.10.5 Inadvertent actuation of QBIS
15.10.6 Inadvertent actuation of algorithm of the primary-to-secondary leak accident management

Following PIEs are categorized as category 3 events:

Increase in heat removal through the secondary circuit
15.1.5 Break of secondary-side pipeline (small leak)

Reactivity and power distribution abnormalities
15.4.2 Uncontrolled withdrawal of one CPS AR
15.4.4 Incorrect loading and operation of fuel assemblies in improper position
15.4.7 Connection of inactive loop without preliminary decrease in power

Decrease in the primary coolant inventory
15.6.1 Inadvertent opening of pressurizer safety valve followed by its failure to seat
15.6.2 SB LOCAs resulting from the break of the primary circuit pipeline with equivalent diameter below 100 mm

Leak into secondary circuit system
15.7.1 Rupture of steam generator heat exchanging tube

Fuel mishandling
15.8.1 Loss of coolant accidents in reactor during shutdown with reactor unsealed and under refuelling
15.8.3 Compensable leak of spent fuel pool facing

Radioactive releases from the systems and equipment of other circuits and systems
15.9.1 Leak or damage of the systems containing liquid radioactive media
15.9.2 Leak of medium from the tank containing radioactive substances
15.9.3 Leak of pipelines in the systems for transporting, storing and processing radwaste containing radioactive gas

Following PIEs are categorized as category 4 events:

Increase in heat removal through the secondary circuit
15.1.6 Break of main steam line

Decrease in heat removal through the secondary circuit
15.2.6 Break of steam generator main feed water line

Decrease in the primary coolant flow rate
15.3.3 Instantaneous seizure or break of one reactor coolant pump set shaft

Reactivity and power distribution abnormalities
15.4.8 Ejection of CPS control rods in case of drive housing rupture

Decrease in the primary coolant inventory
15.6.3 LB LOCAs resulting from the break of the primary circuit pipeline with equivalent diameter exceeding 100 mm, including the break of main coolant pipeline

Leak into secondary circuit system

15.7.2 Primary-to-secondary coolant leak in case of steam generator collector cover lift-off

Fuel mishandling

15.8.4.1 Sticking of the spent FA during refuelling

15.8.4.2 Failures of the equipment of the complex of fuel handling and storage systems, including complete loss of power supply.

15.8.4.3 Decrease in concentration of homogeneous absorber in the spent fuel pool water.

15.8.4.4 Damage of package fastening during transportation of nuclear fuel

15.8.4.5 Drop of transport cask with spent FAs

Example 2

Type of reactor: PWR

Classification of Transients and Accidents

The classification of initiating events is defined by their effect on the RCS. They are categorized according to their expected frequency of occurrence, which provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. Each initiating event is categorized as an anticipated operational occurrence (AOO), a postulated accident (PA), or a beyond design basis event. AOOs, as defined in Appendix A to 10 CFR 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear plant unit. The Standard Review Plan (SRP) presented in NUREG-0800 (Reference 1) refers to AOOs as incidents of moderate frequency (i.e., events that are expected to occur several times during the plant's lifetime) and infrequent events (i.e., events that may occur during the lifetime of the plant). PAs are unanticipated occurrences; they are postulated to occur but not expected to occur during the life of the nuclear plant unit.

AOOs and PAs for this PWR fall into one of the following event types:

- Radioactive release from a subsystem or component.
- Increase in heat removal by the secondary system.
- Decrease in heat removal by the secondary system.
- Decrease in RCS flow rate.
- Reactivity and power distribution anomaly.
- Increase in RCS inventory.
- Decrease in RCS inventory.

For the this reactor, the range of events considered in the safety analysis is developed by considering potential failures in plant systems or operator errors for each initiating event type as defined above. The resulting initiating events are further categorized as either an AOO or PA, depending on expected frequency of occurrence.

Table 15.0-1 Initiating Events provides a list of initiating events analyzed for this reactor, along with the frequency and type categorization for the event.

Table 15.0-1 - Initiating Events

Sheet 1 of 2

Event	Classification
15.0.3 Radioactive Release from Subsystem or Component	**DBA (AOO or PA)**
Failure of small line carrying primary coolant outside containment	DBA (PA)
SG tube failure	DBA (PA)
MSL failure outside containment	DBA (PA)
RCP locked rotor	DBA (PA)
RCCA ejection	DBA (PA)
Fuel handling accident	DBA (PA)
LOCA	DBA (PA)
15.1 Increase in Heat Removal By Secondary System	**AOO or PA**
Decrease in feedwater temperature	AOO
Increase in feedwater flow	AOO
Increase in steam flow	AOO
Inadvertent opening of SG relief or safety valve	AOO
Steam system piping failure	PA[1]
15.2 Decrease in Heat Removal By Secondary System	**AOO or PA**
Loss of external load	AOO
TT	AOO
Loss of condenser vacuum	AOO
Closure of MSIV	AOO
Loss of nonemergency AC power	AOO
Loss of normal feedwater flow	AOO
Feedwater system pipe break	PA[1]
15.3 Decrease in RCS Flow Rate	**AOO or PA**
Partial loss of forced reactor coolant flow	AOO
Complete loss of forced reactor coolant flow	AOO
RCP rotor seizure	PA
RCP shaft break	PA
15.4 Reactivity and Power Distribution Anomaly	**AOO or PA**
Uncontrolled RCCA withdrawal from subcritical or low power startup condition	AOO
Uncontrolled RCCA withdrawal at power	AOO

Sheet 2 of 2

Event	Classification
Single RCCA withdrawal	AOO
RCCA misalignment	AOO
RCCA drop	AOO
Startup of RCP in inactive loop	AOO
Inadvertent decrease in boron concentration in RCS	AOO
Inadvertent loading and operation of fuel assembly in improper position	AOO
RCCA ejection	PA
15.5 Increase in RCS Inventory	**AOO or PA**
Inadvertent operation of ECCS or EBS	AOO
CVCS malfunction that increases reactor coolant inventory	AOO
15.6 Decrease in RCS Inventory	**AOO or PA**
Inadvertent opening of PSRV	AOO
SG tube failure	PA
SBLOCA	PA
LBLOCA	PA

Note:

1. Minor pipe breaks are considered AOOs.

Example 3

Type of reactor: PWR

15.0.1 Classification of Plant Conditions

The ANSI 18.2 (Reference 1) classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

. Condition I: Normal operation and operational transients
. Condition II: Faults of moderate frequency
. Condition III: Infrequent faults
. Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk, and those extreme situations having the potential for the greatest risk should be those least likely to occur. Where applicable, reactor trip and engineered safeguards functioning are assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle.

15.0.1.1 Condition I: Normal Operation and Operational Transients

Condition I occurrences are those that are expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between a plant parameter and the value of that parameter requiring either automatic or manual protective action.

Because Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions that can occur during Condition I operation.

A typical list of Condition I events follows.

Steady-state and Shutdown Operations

See Table 1.1-1 of Chapter 16.

Operation with Permissible Deviations

Various deviations that occur during continued operation as permitted by the plant Technical Specifications are considered in conjunction with other operational modes.

These deviations include the following:

- Operation with components or systems out of service (such as an inoperable rod cluster control assembly [RCCA])
- Leakage from fuel with limited cladding defects
- Excessive radioactivity in the reactor coolant:
    - Fission products
    - Corrosion products
    - Tritium
- Operation with steam generator tube leaks
- Testing

Operational Transients

- Plant heatup and cooldown
- Step load changes (up to +10 percent)
- Ramp load changes (up to 5 percent/minute)
- Load rejection up to and including design full-load rejection transient

15.0.1.2 Condition II: Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault (Condition III or IV events). In addition, Condition II events are not expected to result in fuel rod failures, reactor coolant system failures, or secondary system overpressurization. The following faults are included in this category:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (see subsection 15.1.1)
- Feedwater system malfunctions that result in an increase in feedwater flow (see subsection 15.1.2)
- Excessive increase in secondary steam flow (see subsection 15.1.3)
- Inadvertent opening of a steam generator relief or safety valve (see subsection 15.1.4
- Inadvertent operation of the passive residual heat removal heat exchanger (see subsection 15.1.6)
- Loss of external electrical load (see subsection 15.2.2)
- Turbine trip (see subsection 15.2.3)
- Inadvertent closure of main steam isolation valves (see subsection 15.2.4)
- Loss of condenser vacuum and other events resulting in turbine trip (see subsection 15.2.5)
- Loss of ac power to the station auxiliaries (see subsection 15.2.6)
- Loss of normal feedwater flow (see subsection 15.2.7)
- Partial loss of forced reactor coolant flow (see subsection 15.3.1)
- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup

condition (see subsection 15.4.1)
- Uncontrolled RCCA bank withdrawal at power (see subsection 15.4.2)
- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (see subsection 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (see subsection 15.4.4)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (see subsection 15.4.6)
- Inadvertent operation of the passive core cooling system during power operation (see subsection 15.5.1)
- Chemical and volume control system malfunction that increased reactor coolant inventory (see subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (see subsection 15.6.1)
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (see subsection 15.6.2)

15.0.1.3 Condition II: Infrequent Faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 100. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (see subsection 15.1.5)
- Complete loss of forced reactor coolant flow (see subsection 15.3.2)
- RCCA misalignment (single RCCA withdrawal at full power) (see subsection 15.4.3)
- Inadvertent loading and operation of a fuel assembly in an improper position (see subsection 15.4.7)
- Inadvertent operation of automatic depressurization system (see subsection 15.6.1)
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see subsection 15.6.5)
- Gas waste management system leak or failure (see subsection 15.7.1)
- Liquid waste management system leak or failure (see subsection 15.7.2)
- Release of radioactivity to the environment due to a liquid tank failure (see subsection 15.7.3)
- Spent fuel cask drop accidents (see subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (see subsection 15.1.5)
- Feedwater system pipe break (see subsection 15.2.8)
- Reactor coolant pump shaft seizure (locked rotor) (see subsection 15.3.3)
- Reactor coolant pump shaft break (see subsection 15.3.4)
- Spectrum of RCCA ejection accidents (see subsection 15.4.8)
- Steam generator tube rupture (see subsection 15.6.3)
- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (see subsection 15.6.5)
- Design basis fuel handling accidents (see subsection 15.7.4)