IAEA Safety Assessment Education and Training (SAET) Programme

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

Acceptance criteria Exercise

IAEA

International Atomic Energy Agency

Regulatory requirements on acceptance criteria

Regulatory requirements on acceptance criteria to be used in exercise are defined in IAEA SRS-30, see pages 8 and 9. Download the IAEA SRS-30 from SAN portal or from:

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

During the exercise acceptance criteria for PIE in categories of transient and design basis accidents (DBA) will be reviewed. Below, excerpts for 2 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 SRS-30 document and review the recommended acceptance criteria for transients and DBAs
2. Review the provided information on acceptance criteria for 2 different types of reactors and compare it with the recommendations given in IAEA SRS-30. Focus on:
    a. Suitability of the acceptance criteria toward the PIE category
    b. Completeness of acceptance criteria
    c. Numerical values used for specific acceptance criteria
3. Review the provided calculated results for two categories of PIEs, make statement on completeness of acceptance criteria, identify the values for relevant acceptance criteria and make statement on the acceptance criteria fulfillment
4. Prepare the summary of the review for the presentation

Acceptance criteria of reactor no. 1

Type of reactor: PWR
Rated power: 3000 MWth

Acceptance criteria for anticipated operational occurrences with a frequency of occurrence more than 10^{-2} events per year

1. Pressure in the primary coolant system and SG steam lines will be less than 115 % of the design value

2. The minimum time is available before reaching criticality since the moment the alarm signal arrives that informs the operator of an inadvertent decrease in the boric acid concentration:

   a) during refuelling – 30 min;
   b) during start-up, cold shutdown, hot state and power operation – 15 min

3. The minimum DNBR shall be not less than the assigned value for the assumed correlation with a confidence probability of 95 %. The applied DNBR correlations shall be based on experimental results that correspond to the conditions of core cooling and to the given design of fuel

4. Fuel pellets do not melt even locally (temperature is less than 2540 ºC for “burnt-up” fuel and less than 2840 ºC for “fresh” fuel)

5. Radially-averaged fuel enthalpy shall not exceed 837 J/g UO2 along the axis of any fuel rod

Acceptance criteria for accident conditions 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); and accident conditions with frequency of occurrence ranging from 10^{-4} to 10^{-6} events per year

1. Pressure in the primary coolant system and SG steam lines will be less than 115 % of the design value

2. Instrumentation shall be envisaged to detect possible errors in fuel loading under bringing to power

3. Fuel pellets do not melt even locally (temperature is less than 2840 ºC for “burnt- up” fuel and less than 2540 ºC for “fresh” fuel

4. The following criteria of emergency core cooling shall be met:

a) maximum temperature of fuel rod cladding is less than 1200 °C;
b) local oxidation fraction of fuel rod cladding does not exceed 17 % of the initial cladding thickness;
c) mass of zirconium oxidized is not more than 1 % of its mass in fuel rod claddings

5. Radially-averaged fuel enthalpy shall not exceed 837 J/g UO2 along the axis of any fuel rod

6. The number of fuel rods with DNB shall not exceed 10 % of the total number of fuel rods in the core

7. The number of damaged fuel rods shall not exceed 1 % of the total number of fuel rods in the core

8. The number of damaged fuel rods shall not exceed 10 % of the total number of fuel rods in the core

Results of the analysis no. 1

Increase in primary coolant inventory - malfunction in chemical and volume control system leading to increase in the primary coolant inventory caused by water injection

Category – anticipated operational occurrence

ID	Criterion	Limiting value	Calculated value
1	Pressure in the reactor coolant system and SG steam lines will be less than 115 % of the design value*		Primary side 18.14 MPa Secondary side 8.11 MPa
2	The minimum DNBR shall be not less than the assigned value for the assumed correlation with a confidence probability of 95 %. The applied DNBR correlations shall be based on experimental results that correspond to the conditions of core cooling and to the given design of fuel		1.3
3	Fuel pellets do not melt even locally		1942.5°C
4	Radially-averaged fuel enthalpy shall not exceed limiting value for UO_2 along the axis of any fuel rod		447.2

* design pressure value for primary side is 17.64 MPa and secondary side 7.05

Acceptance criteria of reactor no. 2

Type of reactor: PWR
Rated power: 4500 MWth

Safety Analysis Acceptance Criteria for AOOs

1. Pressure in the reactor coolant and main steam systems maintained below 110% of design value.

2. Fuel cladding integrity is maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit.

3. FCM is precluded by limiting the maximum linear power density.

4. Fuel uniform cladding strain does not exceed one percent.

5. An AOO should not result in a postulated event without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

Safety Analysis Acceptance Criteria for Postulated Accidents

1. Pressure in the RCS and main steam system maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. Fuel cladding integrity is maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed.

3. The release of radioactive material does not result in offsite doses in excess of 10 CFR 100.

4. A PA does not, by itself, cause a consequential loss of function of systems needed to cope with the fault, including those of the RCS and the containment system.

5. For LOCAs, the acceptance criteria of 10 CFR 50.46 also apply.

Results of the analysis no. 2

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

Category – postulated accident

ID	Criterion	Limiting value	Calculated value
1	Pressure in the reactor coolant system and SG steam lines will be less than 115 % of the design value		Primary side below initial value Secondary side 8.88 MPa
2	Fuel pellets do not melt even locally		1889.0°C
3	The following criteria of emergency core cooling shall be met: a) maximum temperature of fuel rod cladding is less than limiting value; b) local oxidation fraction of fuel rod cladding does not exceed limiting value of the initial cladding thickness; c) mass of zirconium oxidized is not more than limiting value of its mass in fuel rod claddings		a) 350.5°C b) No significant oxidation c) No significant zirconium
4	Radially-averaged fuel enthalpy shall not exceed limiting value for UO_2 along the axis of any fuel rod		442.4

* design pressure value for primary side is 17.64 MPa and secondary side 7.05