POST-FUKUSHIMA "STRESS TESTS" OF EUROPEAN NUCLEAR POWER PLANTS – CONTENTS AND FORMAT OF NATIONAL REPORTS

This document is intended to provide guidance for the European Nuclear Regulators on application of ENSREG document ***Annex I, EU "Stress test" specifications***

The guidance is given by way of indication. It is liable to be adjusted during the writing and integration of the report (e.g. to summarize aspects to improve comprehensibility of licensee's explanations). It should be used by the European Nuclear Regulators so that the reports are as homogeneous as possible.

The National Reports shall be written in English and be aimed for full release to the public. They should be detailed enough to give adequate understanding of the robustness of the design but avoid revealing security relevant details. This implies that presenting information on details of systems design and on location and physical protection of equipment should be avoided.

For each chapter, the national report shall present first generic information relevant to all sites (such as regulatory requirements), then recall the main results of the operator analysis and the regulator assessment and conclusions.

It is expected that each regulator follows the numbering provided in the contents below in order to facilitate the peer review. The report length should be between 50 to 200 pages, preferably around 100 pages.

1. General data about the sites and nuclear power plants

1.1. Brief description of the site characteristics

- location (sea, river)
- number of units;
- license holder

1.1.1. Main characteristics of the units

- reactor type;
- thermal power;
- date of first criticality;
- existing spent fuel storage (or shared storage).

1.1.2. Description of the systems for conduction of main safety functions

In this section, all relevant systems should be identified and described, whether they are classified and accordingly qualified as safety systems, or designed for normal operation and classified to non-nuclear safety category. The systems description should include also fixed hook-up points for transportable external power or water supply systems that are planned to be used as last resort during emergencies.

1.2. Significant differences between units

This section is relevant only for sites with multiple NPP units of similar type.

In case some site has units of completely different design (e.g., PWR’s and BWR’s or plants of different generation), design information of each unit is presented separately.

1.3. Use of PSA as part of the safety assessment

Qualitative description of the use of PSA when evaluating the plant safety taking into account the current scope of the analyses. No quantitative values are expected to be quoted.

2. Earthquakes

Both the reactor and spent fuel pools, as well as spent fuel storages at site, are to be considered.

2.1. Design basis

2.1.1. Earthquake against which the plants are designed

Characteristics of the design basis earthquake (DBE)

Level of DBE expressed in terms of maximum horizontal peak ground acceleration (PGA). If no DBE was specified in the original design due to the very low seismicity of the site, PGA that was used to demonstrate the robustness of the as built design.

Methodology used to evaluate the design basis earthquake

Expected frequency of DBE, statistical analysis of historical data, geological information on site, safety margin.

Conclusion on the adequacy of the design basis for the earthquake

Reassessment of the validity of earlier information taking into account the current state-of-the-art knowledge.

2.1.2. Provisions to protect the plants against the design basis earthquake

Identification of systems, structures and components (SSC) that are required for achieving safe shutdown state and are most endangered during an earthquake. Evaluation of their robustness in connection with DBE and assessment of potential safety margin.

Main operating contingencies in case of damage that could be caused by an earthquake and could threaten achieving safe shutdown state.

Protection against indirect effects of the earthquake, for instance

Assessment of potential failures of heavy structures, pressure retaining devices, rotating equipment, or systems containing large amount of liquid that are not designed to withstand DBE and that might threaten heat transfer to ultimate heat sink by mechanical interaction or through internal flood.

Loss of external power supply that could impair the impact of seismically induced internal damage at the plants.

Situation outside the plants, including preventing or delaying access of personnel and equipment to the site.

Other indirect effects (e.g. fire or explosion).

2.1.3. Compliance of the plants with its current licensing basis

Licensee's processes to ensure that plants systems, structures, and components that are needed for achieving safe shutdown after earthquake, or that might cause indirect effects discussed under the previous section remain in operable conditions.

Licensee's processes to ensure that mobile equipment and supplies that are planned to be available after an earthquake are in continuous preparedness to be used.

Potential deviations from licensing basis and actions to address those deviations.

2.2. Evaluation of safety margins

2.2.1. Range of earthquake leading to severe fuel damage

Weak points and cliff edge effects: estimation of PGA above which loss of fundamental safety functions or severe damage to the fuel (in vessel or in fuel storage) becomes unavoidable.

2.2.2. Range of earthquake leading to loss of containment integrity

Estimation of PGA that would result in loss of integrity of the reactor containment.

2.2.3. Earthquake exceeding the design basis earthquake for the plants and consequent flooding exceeding design basis flood

Possibility of external floods caused by an earthquake and potential impacts on the safety of the plants. Evaluation of the geographical factors and the physical possibility of an earthquake to cause an external flood on site, e.g. a dam failure upstream of the river that flows past the site.

2.2.4. Measures which can be envisaged to increase robustness of the plants against earthquakes

Consideration of measures, which could be envisaged to increase plants robustness against seismic phenomena and would enhance plants safety.

3. Flooding

Both the reactor and spent fuel pools, as well as spent fuel storages at site, are to be considered.

3.1. Design basis

3.1.1. Flooding against which the plants are designed

Characteristics of the design basis flood (DBF)

Maximum height of flood postulated in design of the plants and maximum postulated rate of water level rising. If no DBF was postulated, evaluation of flood height that would seriously challenge the function of electrical power systems or the heat transfer to the ultimate heat sink.

Methodology used to evaluate the design basis flood.

Reassessment of the maximum height of flood considered possible on site, in view of the historical data and the best available knowledge on the physical phenomena that have a potential to increase the height of flood. Expected frequency of the DBF and the information used as basis for reassessment.

Conclusion on the adequacy of protection against external flooding

3.1.2. Provisions to protect the plants against the design basis flood

Identification of systems, structures and components (SSC) that are required for achieving and maintaining safe shutdown state and are most endangered when flood is increasing.

Main design and construction provisions to prevent flood impact to the plants.

Main operating provisions to prevent flood impact to the plants.

Situation outside the plants, including preventing or delaying access of personnel and equipment to the site.

3.1.3. Plants compliance with its current licensing basis

Licensee's processes to ensure that plants systems, structures, and components that are needed for achieving and maintaining the safe shutdown state, as well as systems and structures designed for flood protection remain in operable condition.

Licensee's processes to ensure that mobile equipment and supplies that are planned for use in connection with flooding are in continuous preparedness to be used.

Potential deviations from licensing basis and actions to address those deviations.

3.2. Evaluation of safety margins

3.2.1. Estimation of safety margin against flooding

Estimation of difference between maximum height of flood considered possible on site and the height of flood that would seriously challenge the safety systems, which are essential for heat transfer from the reactor and the spent fuel to ultimate heat sink.

3.2.2. Measures which can be envisaged to increase robustness of the plants against flooding.

Consideration of measures, which could be envisaged to increase plants robustness against flooding and would enhance plants safety.

4. Extreme weather conditions

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

Verification of weather conditions that were used as design basis for various plants systems, structures and components: maximum temperature, minimum temperature, various type of storms, heavy rainfall, high winds, etc.

Postulation of proper specifications for extreme weather conditions if not included in the original design basis.

Assessment of the expected frequency of the originally postulated or the redefined design basis conditions.

Consideration of potential combination of weather conditions.

Conclusion on the adequacy of protection against extreme weather conditions

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

Analysis of potential impact of different extreme weather conditions to the reliable operation of the safety systems, which are essential for heat transfer from the reactor and the spent fuel to ultimate heat sink.

Estimation of difference between the design basis conditions and the cliff edge type limits, i.e. limits that would seriously challenge the reliability of heat transfer.

4.2.2. Measures which can be envisaged to increase robustness of the plants against extreme weather conditions

Consideration of measures, which could be envisaged to increase plants robustness against extreme weather conditions and would enhance plants safety.

5. Loss of electrical power and loss of ultimate heat sink

For writing chapter 5, it is suggested that the emphasis is in consecutive measures that could be attempted to provide necessary power supply and decay heat removal from the reactor and from the spent fuel.

Chapter 5 should focus on prevention of severe damage of the reactor and of the spent fuel, including all last resort means and evaluation of time available to prevent severe damage in various circumstances. As opposite, the Chapter 6 should focus on mitigation, i.e. the actions to be taken after severe reactor or spent fuel damage as needed to prevent large radioactive releases. Main focus in Chapter 6 should thus be in protection of containment integrity.

5.1. Loss of electrical power

All offsite electric power supply to the site is lost. The offsite power should be assumed to be lost for several days. The site is isolated from delivery of heavy material for 72 hours by road, rail or waterways. Portable light equipment can arrive to the site from other locations after the first 24 hours.

5.1.1. Loss of off-site power

Design provisions taking into account this situation: normal back-up AC power sources provided, capacity and preparedness to take them in operation, Dependence on the functions of other reactors on the same site. Robustness of the provisions in connection with seism and flooding.

Autonomy of the on-site power sources and provisions taken to prolong the service time of on-site AC power supply

5.1.2. Loss of off-site power and loss of the ordinary back-up AC power source

Design provisions taking into account this situation: diverse permanently installed AC power sources and/or means to timely provide other diverse AC power sources, capacity and preparedness to take them in operation. Robustness of the provisions in connection with seismic events and flooding.

Battery capacity, duration and possibilities to recharge batteries

5.1.3. Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

Battery capacity, duration and possibilities to recharge batteries in this situation

Actions foreseen to arrange exceptional AC power supply from transportable or dedicated off-site source

Competence of shift staff to make necessary electrical connections and time needed for those actions. Time needed by experts to make the necessary connections.

Time available to provide AC power and to restore core and spent fuel pool cooling before fuel damage: consideration of various examples of time delay from reactor shutdown and loss of normal reactor core cooling condition (e.g., start of water loss from the primary circuit).

5.1.4. Conclusion on the adequacy of protection against loss of electrical power.

5.1.5. Measures which can be envisaged to increase robustness of the plants in case of loss of electrical power

5.2. Loss of the decay heat removal capability/ultimate heat sink

The connection with the primary ultimate heat sink for all safety and non safety functions is lost. The site is isolated from delivery of heavy material for 72 hours by road, rail or waterways. Portable light equipment can arrive to the site from other locations after the first 24 hours.

5.2.1. Design provisions to prevent the loss of the primary ultimate heat sink, such as alternative inlets for sea water or systems to protect main water inlet from blocking.

Robustness of the provisions in connection with seism and flooding.

5.2.2. Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)

Availability of an alternate heat sink, dependence on the functions of other reactors on the same site.

Possible time constraints for availability of alternate heat sink and possibilities to increase the available time.

5.2.3. Loss of the primary ultimate heat sink and the alternate heat sink

External actions foreseen to prevent fuel degradation.

Time available to recover one of the lost heat sinks or to initiate external actions and to restore core and spent fuel pool cooling before fuel damage: consideration of various examples of time delay from reactor shutdown to loss of normal reactor core and spent fuel pool cooling condition (e.g., start of water loss from the primary circuit).

5.2.4. Conclusion on the adequacy of protection against loss of ultimate heat sink

5.2.5. Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink

5.3. Loss of the primary ultimate heat sink, combined with station black out (see stress tests specifications).

5.3.1. Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g., start of water loss from the primary circuit).

5.3.2. External actions foreseen to prevent fuel degradation.

5.3.3. Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out

6. Severe accident management

6.1. Organization and arrangements of the licensee to manage accidents

Section 6.1 should cover organization and arrangements for managing all type of accidents, starting from design basis accidents where the plants can be brought to safe shutdown without any significant nuclear fuel damage and up to severe accidents involving core meltdown or damage of the spent nuclear fuel in the storage pool.

6.1.1. Organisation of the licensee to manage the accident

Staffing and shift management in normal operation

Measures taken to enable optimum intervention by personnel

Use of off-site technical support for accident management

Dependence on the functions of other reactors on the same site

Procedures, training and exercises.

Plans for strengthening the site organisation for accident management

6.1.2. Possibility to use existing equipment

Provisions to use mobile devices (availability of such devices, time to bring them on site and put them in operation)

Provisions for and management of supplies (fuel for diesel generators, water, etc.)

Management of radioactive releases, provisions to limit them

Communication and information systems (internal and external).

6.1.3. Evaluation of factors that may impede accident management and respective contingencies

Extensive destruction of infrastructure or flooding around the installation that hinders access to the site

Loss of communication facilities / systems

Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation

Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident

Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

Unavailability of power supply

Potential failure of instrumentation

Potential effects from the other neighbouring installations at site, including considerations of restricted availability of trained staff to deal with multi-unit, extended accidents.

6.1.4. Conclusion on the adequacy of organisational issues for accident management

6.1.5. Measures which can be envisaged to enhance accident management capabilities

6.2. Accident management measures in place at the various stages of a scenario of loss of the core cooling function

6.2.1. Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes (including last resorts to prevent fuel damage)

6.2.2. After occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

6.2.3. After failure of the reactor pressure vessel/a number of pressure tubes

6.3. Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

6.3.1. Elimination of fuel damage / meltdown in high pressure

Design provisions

Operational provisions

6.3.2. Management of hydrogen risks inside the containment

Design provisions, including consideration of adequacy in view of hydrogen production rate and amount

Operational provisions

6.3.3. Prevention of overpressure of the containment

Design provisions, including means to restrict radioactive releases if prevention of overpressure requires steam / gas relief from containment

Operational and organisational provisions

6.3.4. Prevention of re-criticality

Design provisions

Operational provisions

6.3.5. Prevention of basemat melt through

Potential design arrangements for retention of the corium in the pressure vessel

Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

Cliff edge effects related to time delay between reactor shutdown and core meltdown

6.3.6. Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

Design provisions

Operational provisions

6.3.7. Measuring and control instrumentation needed for protecting containment integrity

6.3.8. Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

6.3.9. Conclusion on the adequacy of severe accident management systems for protection of containment integrity

6.3.10. Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

6.4. Accident management measures to restrict the radioactive releases

6.4.1. Radioactive releases after loss of containment integrity

Design provisions

Operational provisions

6.4.2. Accident management after uncovering of the top of fuel in the fuel pool

Hydrogen management

Providing adequate shielding against radiation

Restricting releases after severe damage of spent fuel in the fuel storage pools

Instrumentation needed to monitor the spent fuel state and to manage the accident

Availability and habitability of the control room

6.4.3. Conclusion on the adequacy of measures to restrict the radioactive releases

7. General conclusion

7.1. Key provisions enhancing robustness (already implemented)

Safety margins identified and their significance. Any provisions or good practices that lead to enhance the robustness of the plants, implemented for example following continuous improvement process or PSR.

7.2. Safety issues

Shortfalls, if any, and cliff edge effects identified and their significance

7.3. Potential safety improvements and further work forecasted

The regulator is expected to give an overview of the studies or improvements that have been decided after stress tests and an indication of corresponding the time scale.