Trawsfynydd:
Response to EU Stress Tests following the Events at Fukushima, Japan

Following the nuclear accident at Fukushima in Japan, the European Union agreed on assessments for all of its 143 nuclear power plants, based on a set of common criteria. These criteria have been developed by ENSREG (the European Nuclear Safety Regulators Group) and have become known as 'Stress Tests'.

In response to the Stress Tests, operators of UK nuclear power plants have reviewed the resilience of their plants to extreme situations, in particular the loss of safety functions however caused, including the loss of electrical power or loss of ultimate heat sink for heat removal from the reactor or spent fuel storage areas.

This report details the results of the Stress Test for Trawsfynydd Site. It has been submitted to the Office for Nuclear Regulation (an agency of the Health and Safety Executive) who will review all UK submissions and prepare a summary national report. This will be reviewed by ENSREG who will report to the European Council in June 2012.

Issued by

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0 Executive Summary

This report is the response from Trawsfynydd Site to the ENSREG Stress Tests following the events at Fukushima, Japan in March 2011.

Trawsfynydd site comprises the remaining structures of Trawsfynydd nuclear power station, with two Magnox, gas cooled, graphite moderated, natural uranium reactors, a cooling ponds building and various ancillary buildings.

The reactors are defuelled and there is no fuel remaining on site, therefore there are no requirements for reactivity control, cooling or ultimate heat sinks. The reactors have been placed into a benign "Safestore" condition, which means that they require minimal services to maintain their long-term physical integrity and there are no safety demands on site services. The fuel storage ponds on site contain no fuel and have been drained and are being decommissioned.

Intermediate level waste (ILW) and low level waste (LLW) are contained partly in purpose built storage facilities and partly in redundant process facilities. It was recognised (prior to the Fukushima event) that the ILW stored in underground concrete vaults is the only radioactive material that would be at significant risk from site-wide flooding from the adjacent lake. Assessments have concluded that the vaults are protected against the 10^{-4} per annum design basis flood (from the lake adjacent to the site and as a direct result of dam failure and water overtopping) by the topography of the lake dam in relation to the site.

A Periodic Safety Review of the site was completed in 2011 and the safety cases for the reactor Safestores and all the radioactive waste management facilities have been reviewed against modern standards. It is concluded that the facilities are robust to the challenges of extreme external hazards used in the analyses and retain their basic integrity for somewhat more severe events and combinations.

As a result of a recent survey it has been revealed that the Reactor Building structures do not meet the current wind code requirements for a 2×10^{-2}/y event. A programme of remedial strengthening works has been instigated which will progressively reduce the risk.

A series of workshops has been held to identify potential measures to enhance resilience in the event of external hazards or severe accidents, and those being considered for implementation are listed in Table 1. The site will also be supported by enhancements proposed for central emergency support.

1 General data about site/plant

1.1 Brief description of the site characteristics

- location (sea, river)[1]
- number of units;
- license holder

Sited in the Snowdonia National Park in Gwynedd, North Wales, UK adjacent to Trawsfynydd Lake (Llyn Trawsfynydd), circa 195m above sea level, Trawsfynydd Power Station was commissioned in 1965. The site consisted of twin Magnox reactors, a cooling pond building and a multitude of ancillary buildings. Generation ceased at Trawsfynydd in February 1991 and the Station was formally shut down in July 1993. All irradiated fuel was removed from site by August 1995. The site is currently being decommissioned before entry into a Care and Maintenance state that will eventually lead to Final Site Clearance.

Magnox Limited is the Site Licence holder for the Trawsfynydd nuclear licensed site.

1.2 Main characteristics of the unit

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

<u>Reactor 1</u>

Reactor 1 is a defuelled Magnox reactor. When operating it contained natural metallic uranium fuel in cans of magnesium/aluminium alloy, known as Magnox. It had a thermal capacity of 850MW. The core was cooled by forced circulation of CO_2 gas through the core, transferring heat to boilers supplying steam to turbine generators. The reactor core is contained in a spherical steel pressure vessel, which was surrounded by six vertical boilers.

It is currently defuelled with an air atmosphere at atmospheric pressure with no cooling.

<u>Reactor 2.</u>

Reactor 2 is the same design as Reactor 1 and is also defuelled with an air atmosphere at atmospheric pressure with no cooling.

[1] Text and headings which are in a smaller font are relevant extracts from the ENSREG Stress Test documentation and not part of the Stress Test response.

Irradiated Fuel Cooling Ponds

Ponds provided cooling and shielding of irradiated fuel discharged from Reactors 1 and 2 before they were transferred to fuel flasks for transport to Sellafield. The ponds no longer contain fuel, are drained and are currently in the process of being decommissioned.

Radioactive Waste Facilities

There is no high level waste (HLW) on site. Intermediate level waste (ILW) and low level waste (LLW) are contained in either the original process buildings or in purpose built storage facilities:

The Reactor Buildings – Each of the two reactor buildings contains a spherical steel reactor pressure vessel containing the defuelled irradiated core and supporting structure. The buildings also contain the partly dismantled gas circuits, the Miscellaneous Activated Components (MAC) vaults and the Mortuary Holes. The MAC vaults had contained miscellaneous activated components (MAC) but have now been emptied. The Mortuary holes are steel tubes located within the biological shield and accessed from the pile cap. They contain redundant activated plant items which are intended to be stored there throughout the Care and Maintenance phase. Additionally, the Gas Circulator sub-basements have been converted to provide interim storage for packages of solid ILW encapsulated in grout and contained in standard Radioactive Waste Management Directorate (RWMD) approved containers within reinforced concrete overpacks; these packages are being progressively transferred to the new ILW store.

Main Active Waste Vaults – these are a series of nine below-ground vaults constructed in reinforced concrete. The vaults are approximately 4.8m deep with covering slabs of either 560mm or 838mm thick reinforced concrete. The vaults contain some loose contaminated items and various packages and drums containing a variety of ILW and LLW.

Magnox Debris Handling and Storage Facility (MDHSF) – the facility comprises six above-ground reinforced concrete storage cells. The external walls are 800mm thick, with partition walls between the cells of 300mm-600mm thickness. Constructed to provide additional Fuel Element Debris (FED) storage capability when the FED Vaults were near capacity, the facility now stores drums of compacted Magnox FED.

Sludge and Resin tanks – the original Main Sludge Vault (MSV) and Resin Vault 1 (RV1) are partially below-ground mild steel tanks contained within reinforced concrete cells which provide secondary containment and radiation shielding. The MSV contains sludge from the cooling ponds and the Active Effluent Treatment Plant (AETP). RV1 contained spent ion exchange resins from the effluent treatment process; it has now been emptied of resin. Two further Resin Vaults, RV2 and RV3, were constructed to provide additional resin storage facilities. These are stainless steel tanks, again contained within reinforced concrete secondary containment cells, which are wholly below-ground. They are located within a separate building.

Pond North Void (PNV) – the PNV is a closed box extension of the cooling pond storage bay and is constructed in the same manner. Originally filled only with water to provide radiation shielding for the irradiated fuel transfer tubes, it is now used for storage of ILW.

Magnox Storage Vaults – these below-ground vaults are rectangular concrete structures of wall and floor thickness 762mm with partition walls of 228mm thickness. The vaults contain FED which comprises pieces of Magnox cladding stripped from the irradiated fuel elements before they were despatched to Sellafield for reprocessing.

Resin Plant and Drum Stores – this plant was built to condition radioactive waste into a solid form suitable for long term storage or disposal. It consists of a Resin encapsulation plant and a shielded storage facility for the drums of encapsulated waste. An additional storage facility for the drums of encapsulated waste was established in a converted workshop. These waste stores will be progressively emptied as their contents are transferred to the new ILW store.

The Active Effluent Treatment Plant (AETP) – this operational plant is provided for the treatment of all liquid effluent that arises from Site operation and the waste facilities. It is shortly to be decommissioned.

Irradiated Fuel Cooling Ponds – this facility was designed and operated for the storage/cooling/shielding of irradiated fuel elements (that had been discharged from the reactor core) until shipped off site for reprocessing. Now that all irradiated fuel has been removed from site, this facility has been drained and is currently being decommissioned.

ILW Store – this is a new purpose-built concrete shielded store for ILW. The store provides safe passive storage for all conditioned ILW (stored in RWMD approved storage containers) for the Care and Maintenance period until despatch off-site for long term storage or disposal at a nationally designated repository. The packages that are stored within the facility are designed to survive the design basis seismic event even if the structure of the building fails.

1.3 Systems for providing or supporting main safety functions

In this chapter, 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.

1.3.1 Reactivity control

Systems that are planned to ensure sub-criticality of the reactor core in all shut down conditions, and sub-criticality of spent fuel in all potential storage conditions. Report should give a thorough understanding of available means to ensure that there is adequate amount of boron or other respective neutron absorber in the coolant in all circumstances, also including the situations after a severe damage of the reactor or the spent fuel.

As there is no fuel present in the reactors no reactivity control is required. The control rods remain in the core and will be disposed of at final site clearance.

1.3.2 Heat transfer from reactor to the ultimate heat sink

1.3.2.1 All existing heat transfer means / chains from the reactor to the primary heat sink (e.g., sea water) and to the secondary heat sinks (e.g., atmosphere or district heating system) in different reactor shut down conditions: hot shut down, cooling from hot to cold shut down, cold shut down with closed primary circuit, and cold shut down with open primary circuit.

1.3.2.2 Lay out information on the heat transfer chains: routing of redundant and diverse heat transfer piping and location of the main equipment. Physical protection of equipment from the internal and external threats.

1.3.2.3 Possible time constraints for availability of different heat transfer chains, and possibilities to extend the respective times by external measures (e.g., running out of a water storage and possibilities to refill this storage).

1.3.2.4 AC power sources and batteries that could provide the necessary power to each chain (e.g., for driving of pumps and valves, for controlling the systems operation).

1.3.2.5 Need and method of cooling equipment that belong to a certain heat transfer chain; special emphasis should be given to verifying true diversity of alternative heat transfer chains (e.g., air cooling, cooling with water from separate sources, potential constraints for providing respective coolant).

As there is no fuel and therefore no heat generating source in the reactors, no heat transfer is required.

1.3.3 Heat transfer from spent fuel pools to the ultimate heat sink

1.3.3.1 All existing heat transfer means / chains from the spent fuel pools to the primary heat sink (e.g., sea water) and to the secondary heat sinks (e.g., atmosphere or district heating system).

1.3.3.2 Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment as explained under 1.3.2.

As there is no fuel present in the spent fuel ponds and therefore no heat generating source no heat transfer is required.

1.3.4 Heat transfer from the reactor containment to the ultimate heat sink

In the Magnox design, the reactor vessel provides the reactor containment (the covering reactor building providing protection of plant from the environment). Cooling of the reactor internals and the pressure vessel is addressed in Section 1.3.2.

1.3.4.1 All existing heat transfer means / chains from the containment to the primary heat sink (e.g., sea water) and to the secondary heat sinks (e.g., atmosphere or district heating system).

1.3.4.2 Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment as explained under 1.3.2.

There is no reactor containment and hence no need for heat transfer

1.3.5 AC power supply

1.3.5.1 Off-site power supply

1.3.5.1.1 Information on reliability of off-site power supply: historical data at least from power cuts and their durations during the plant lifetime.

1.3.5.1.2 Connections of the plant with external power grids: transmission line and potential earth cable routings with their connection points, physical protection, and design against internal and external hazards.

The site electrical supplies are derived from an adjacent 275/400kV sub-station. Supplies to this sub-station are derived from the National Grid via three grid lines.

1.3.5.2 Power distribution inside the plant

1.3.5.2.1 Main cable routings and power distribution switchboards.

1.3.5.2.2 Lay-out, location, and physical protection against internal and external hazards.

The on-site electrical distribution system is via a single 11kV main board feeding two 3.3kV boards which then distribute to various 415V three phase boards situated across the site. The facility safety cases allow for the loss of off-site power for approximately three days. There are, however, no significant consequences for longer periods as the only safety related systems that require electrical supplies are associated with monitoring for hydrogen in the magnox vaults, seismic monitoring and environmental conditions in the reactor vessels. These can be supplied from temporary or standby sources.

1.3.5.3 Main ordinary on-site source for back-up power supply

1.3.5.3.1 On-site sources that serve as first back-up if offsite power is lost.

1.3.5.3.2 Redundancy, separation of redundant sources by structures or distance, and their physical protection against internal and external hazards.

1.3.5.3.3 Time constraints for availability of these sources and external measures to extend the time of use (e.g., fuel tank capacity).

In the event of a total grid loss, standby supplies are provided via diesel generators which will support essential demand. A minimum safety related supply of diesel stored on site would enable circa two weeks of generating capacity without external replenishment. DC electrical supplies (240V and 110V) for the distribution boards are sourced by local Uninterruptible Power Supplies (UPS) of capacity ranging between 8 and 24 hours. The emergency lighting system is provided via battery backed light units.

1.3.5.4 Diverse permanently installed on-site sources for back-up power supply

1.3.5.4.1 All diverse sources that can be used for the same tasks as the main back-up sources, or for more limited dedicated purposes (e.g., for decay heat removal from reactor when the primary system is intact, for operation of systems that protect containment integrity after core meltdown).

1.3.5.4.2 Respective information on location, physical protection and time constraints as explained under 1.3.5.3.

There is a local area network Ring Main Unit positioned adjacent to the 11kV main board, with a cable in place, but not connected at either end. This could be utilised if a long term local National Grid disruption is foreseen.

1.3.5.5 Other power sources that are planned and kept in preparedness for use as last resort means to prevent a serious accident damaging reactor or spent fuel.

1.3.5.5.1 Potential dedicated connections to neighbouring units or to nearby other power plants.

1.3.5.5.2 Possibilities to hook-up transportable power sources to supply certain safety systems.

1.3.5.5.3 Information on each power source: power capacity, voltage level and other relevant constraints.

1.3.5.5.4 Preparedness to take the source in use: need for special personnel, procedures and training, connection time, contract arrangements if not in ownership of the Licensee, vulnerability of source and its connection to external hazards and weather conditions.

This is not applicable for Trawsfynydd Site as both reactors are permanently shutdown and defuelled and the irradiated fuel storage ponds have been permanently emptied of fuel.

1.3.6 Batteries for DC power supply

1.3.6.1 Description of separate battery banks that could be used to supply safety relevant consumers: capacity and time to exhaust batteries in different operational situations.

1.3.6.2 Consumers served by each battery bank: driving of valve motors, control systems, measuring devices, etc.

1.3.6.3 Physical location and separation of battery banks and their protection from internal and external hazards.

1.3.6.4 Alternative possibilities for recharging each battery bank.

Uninterruptible Power Supplies (UPS) are provided on Security, Fire Detection, Voice Alarm (Public Address System), IT Systems and Site Alarm Panel.

1.4 Significant differences between units

This chapter 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.

There are no fundamental significant differences between the reactors – only those which relate to the different stage of decommissioning.

1.5 Scope and main results of Probabilistic Safety Assessments

Scope of the PSA is explained both for level 1 addressing core meltdown frequency and for level 2 addressing frequency of large radioactive release as consequence of containment failure. At each level, and depending on the scope of the existing PSA, the results and respective risk contributions are presented for different initiating events such as random internal equipment failures, fires, internal and external floods, extreme weather conditions, seismic hazards. Information is presented also on PSA's conducted for different initiating conditions: full power, small power, or shut down.

The safety cases for each of the waste management facilities do not include a PSA. The safety cases are based upon a fault schedule, a calculation of the radiological consequences and a calculation of the associated risk, which are shown to be consistent with national and international standards and are as low as reasonably practicable (ALARP).

2 Earthquakes

2.1 Design basis

2.1.1 Earthquake against which the plant is designed

2.1.1.1 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.

Seismic hazards were not included within the original design basis. The capability of the site to withstand seismic events was evaluated as part of the Long Term Safety Review (LTSR) completed in 1991, around the time when the station ceased generation. The key structures, systems and components required to resist earthquake loading were assessed against the Principia Mechanica Limited (PML) hard site United Kingdom (UK) design response spectrum anchored to a horizontal zero period acceleration of 0.1g.

2.1.1.2 Methodology used to evaluate the design basis earthquake

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

No formal site-specific seismic hazard study has been undertaken for the Trawsfynydd site. Rather, the PML UK hard site design response spectra anchored to an appropriate horizontal free-field zero period acceleration have been used for assessment and design of key structures, systems and components.

The PML UK design response spectra are piece-wise linear (on a standard tripartite plot) response spectra derived by statistical analysis of strong motion earthquake records from elsewhere in the world conforming to the profile of expected UK events. This is necessitated by a lack of suitable UK-specific strong motion records. These design spectra may be anchored to any zero period acceleration.

For the purpose of defining the assessment basis event for Trawsfynydd during the LTSR the spectrum was anchored to a horizontal zero period acceleration of 0.1g in recognition of the international regulatory significance of that value. The effective exceedance frequency of that assessment basis seismic demand was argued, at the time of the assessment, to be 1.1×10^{-4} per annum for the majority of plant items.

An approximate probabilistic seismic hazard study completed for the Trawsfynydd site (based on a comprehensive zonation model developed for the Wylfa site, which is approximately 80km from Trawsfynydd) suggests that a peak ground acceleration of 0.1g has an annual probability of exceedance of around 7×10^{-4}. However, for most structures, systems and components, seismic capability is determined by seismic loading at moderate to low frequencies (<10Hz) rather than the high frequencies that determine the peak ground acceleration. At such moderate to low frequencies the ratio between spectral acceleration and peak ground acceleration exhibited by the PML design spectra is significantly higher than that exhibited by typical UK uniform risk spectra. Thus the expected exceedance frequency of the 0.1g PML spectra within that lower frequency range is judged to be consistent with that (1.1×10^{-4} per annum) asserted at the time of the LTSR assessments.

2.1.1.3 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.

The UK as a whole is a region of relatively low-level and diffuse seismic activity. Examination of the pattern of historical UK seismicity indicates that Trawsfynydd is situated in a region of moderate to high earthquake activity by UK standards.

The assessment basis seismic demand for existing structures has not been evaluated by modern probabilistic methods. Notwithstanding that potential shortcoming, it is considered that meeting the demand level employed (0.1g PML hard site spectra) is sufficient to give assurance of adequate seismic robustness. This is particularly so given the relatively low level of radiological hazard posed by seismically-induced failure of those structures, systems and components that have been assessed against that demand.

2.1.2 Provisions to protect the plant against the design basis earthquake

2.1.2.1 Systems Structures and Components (SSCs)

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

As the reactors and the pond are defuelled there are no systems or components providing protection to produce a safe shutdown state.

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

As the reactors and the pond are defuelled there are no systems or components providing protection to produce a safe shutdown state.

2.1.2.3 Protection against indirect effects of the earthquake

2.1.2.3.1 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.

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

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

2.1.2.3.4 Other indirect effects (e.g. fire, explosion).

The indirect effects from failure of non-seismically qualified structures are shown in the facility safety cases to be acceptable. The seismic safety case for the design basis earthquake assumes that the earthquake causes an immediate loss of all incoming electrical power supplies to the site. No reliance is placed on restoration of those supplies for maintenance of essential safety functions. Loss of external power supplies is also considered in Section 5 below. The only other indirect effect could be a delay in personnel getting to site. This would only affect monitoring arrangements and would not affect the safety case.

The 2001 and 2011 Periodic Safety Reviews (PSRs) revisited the seismic safety case but no amendments or additions to the case were made as a result. It was however noted that with the removal from site of the irradiated fuel from the reactors by August 1995, the hazard to the public and site personnel from activities on site had been substantially reduced.

In the period since the last PSR, safety cases have been prepared for the Reactor Buildings Safestore (RBS), the new ILW store and the North and South Magnox storage vaults.

The cases for the RBS and the ILW store have drawn on the LTSR work and conclude that the seismic hazard would pose little threat to engineered structures such as those on Site and therefore presents no significant risk to nuclear safety. Additionally, the hazard assessment for the RBS safety case has assessed the potential dose that could arise from a total loss of containment which gives rise to an airborne release. This assessment concludes that a lifetime dose to a member of the public of 0.075mSv would result from such an event. Such levels of dose are Broadly Acceptable at a return frequency of 10^{-4} per annum and it is judged from the foregoing that the risks posed by Seismic activity are ALARP.

Taking account of all the storage facilities on site accumulations of ILW are well protected from external threats by the significant civil structures that are provided for radiation shielding. The exception is the Resin Drum Store where the ILW has been encapsulated into an inert matrix and packaged into discrete drums. Protection is provided by inert passive means with the only active equipment installed for monitoring purposes; failure of this equipment would not give rise to a release of radioactive material and time would be available for the repair and reinstatement of the monitoring equipment should it fail.

2.1.3 Compliance of the plant with its current licensing basis

2.1.3.1 Processes to ensure SSCs remain in faultless condition

Licensee's processes to ensure that plant systems, structures, and components that are needed for achieving safe shut down after earthquake, or that might cause indirect effects discussed under 2.1.2.3 remain in faultless condition.

The plant is subject to routine maintenance, inspection and testing as required by the Nuclear Maintenance Schedule, which lists those activities that are necessary to support the ongoing site safety case. This is implemented in accordance with MCP (Management Control Procedure) 19 “Management of Maintenance Work” and MCP 13 “Surveillance and Routine Testing of Plant Items and Systems”. Specific procedures include S-268 "Inspection and Assessment of Nuclear Safety Related Civil Structures to Comply with Site Licence Condition 28", whose scope includes all significant civil structures and specifically includes structures claimed for seismic support. Any changes to the plant or the safety case are controlled via MCP 99.

The ability of the facilities to continue to meet their integrity requirements is demonstrated by civil inspections in accordance with Licence Condition 28 arrangements. Planned System Engineering walkdowns are conducted and reported to a defined programme.

At 10-yearly intervals, and in response to significant operating events, the safety of the plant is reviewed in a Periodic Safety Review (PSR). This reviews the plant against modern standards, operating experience and the effect of ageing. A PSR has recently been completed and identified enhancements are being implemented.

It has recently been found that the Reactor Building structures do not meet the current standards for resistance to wind loadings. Consequently, it is possible that extreme winds or other hazards could lead to a failure of part of the structure and possible impact on adjacent buildings. The resulting risks to the public have been shown to be above the target level but still below the statutory limit. A programme of remedial strengthening works has been instigated which will progressively reduce the risk.

2.1.3.2 Processes for mobile equipment and supplies

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.

In addition as part of the emergency arrangements, provision is made for off-site surveys of potential radioactive contamination using local resources (up to 15km) and the resources of other nuclear sites and the Ministry of Defence (MoD (out to 40km)).

2.1.3.3 Potential deviations from licensing basis

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

There are no potential deviations from the licensing basis.

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 that would result in damage to the weakest part of heat transfer chain, and consequently cause a situation where the reactor integrity or spent fuel integrity would be seriously challenged.

There is no fuel present on site so this is not applicable.

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.

There is no containment structure. The integrity of the radioactive waste facilities is considered in Section 2.1.2.

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

Possibility of external floods caused by an earthquake and potential impacts on the safety of the plant. 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.

The potential for and consequences of flooding from sources external to the Site and its effects on nuclear safety were extensively explored for the LTSR in 1991, assessed by a hydrogeology report by consultants in September 2000, and further assessed by the 2001 PSR. It was concluded that, although overtopping of the adjacent dams was possible during the Predicted Maximum Flood, flooding from Llyn Trawsfynydd was not a threat to nuclear safety due to the topography of the surrounding land which would route the water away from the nuclear safety-related structures.

Complete failure of the dam has also been considered; it is not a threat to nuclear safety due to the topography of the surrounding land which would route the water away from the nuclear safety-related structures. It would have the potential to render inoperative the site grid supplies but would not disable required back-up plant.

2.2.4 Potential need to increase robustness of the plant against earthquakes

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

No measures have been identified as necessary to increase the robustness of the facilities against earthquakes other than the already identified programme of work to strengthen the reactor building structures.

3 Flooding

3.1 Design basis

3.1.1 Flooding against which the plant is designed

3.1.1.1 Characteristics of the design basis flood (DBF)

Maximum height of flood postulated in design of the plant 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.

The potential for and consequences of flooding from sources external to the Site and its effects on nuclear safety were extensively explored for the LTSR in 1991, assessed by a hydrogeology report by consultants in September 2000, and further assessed by the 2001 PSR. It was concluded that, although overtopping of the adjacent dams was possible during the Predicted Maximum Flood (based on an event based probability of 10^{-4}), flooding from Llyn Trawsfynydd was not a threat to nuclear safety due to the topography of the surrounding land which would route the water away from the nuclear safety-related structures. It is noted that the lake levels are self limiting due to the retaining dams.

During the subsequent Periodic Safety Reviews (PSR) in 2001 and 2011 the safety case was revisited but no changes or additions to the case were made. Over the life of the station there have been no recorded flooding events of any consequence.

3.1.1.2 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.

It is judged that because the lake level is self-limiting by the retaining dams, the risks from off-site flooding will remain insignificant and ALARP. Complete failure of the dam adjacent to the site has been assessed and due to site topography such an event has been shown to have no effect on the ILW storage area.

3.1.1.3 Conclusion on the adequacy of protection against external flooding

This section addresses the adequacy of the design basis for flooding. The evaluation of safety margins against flooding is addressed in section 3.2.

In the period since the 2001 PSR, safety cases have been prepared for the Reactor Safestore Buildings, the new ILW store and the North and South Magnox Storage vaults. These cases have drawn on both the LTSR work and the Comprehensive Radioactive Waste safety case and all conclude that flooding from external sources presents no significant risk to nuclear safety.

3.1.2 Provisions to protect the plant against the design basis flood

3.1.2.1 Systems Structures and Components (SSCs)

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

The key structures that provide protection against a DBF are the containment structures of the facilities that contain radioactive waste. Components that could provide paths for flood water to enter the facilities are either at a height that are not affected by the DBF or are engineered to preclude water ingress.

3.1.2.2 Main design and construction provisions

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

The main protection of the facilities against flooding is provided by the topography of the surrounding land which would route the water away from the nuclear safety-related structures.

Another reservoir in close proximity to the site is Llyn Celyn that is both a large body of water and at a ground level higher than the site. However, the topography of the area would result in water resulting from dam failure flowing in a direction well away from the site.

The site’s height above sea level (195m) prevents any impact from a Tsunami or sea flood event.

3.1.2.3 Main operating provisions

Main operating provisions to prevent flood impact to the plant.

There are currently no provisions to prevent the flood impact identified.

Complete failure of the dam would disable the site grid supplies but would not render inoperable the required back up plant.

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

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

There are no immediate consequences of delayed access to site as all operations can be safety terminated locally. Equipment and personnel from off site are not required to prevent a compromise to nuclear safety as all irradiated fuel has been removed from site.

3.1.3 Plant compliance with its current licensing basis

3.1.3.1 Processes to ensure SSCs remain in faultless condition

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

The plant is subject to routine maintenance, inspection and testing as required by the Nuclear Maintenance Schedule, which lists those activities that are necessary to support the ongoing site safety case. This is implemented in accordance with MCP 19 “Management of Maintenance Work” and MCP 13 “Surveillance and Routine Testing of Plant Items and Systems”. Any changes to the plant or the safety case are controlled via MCP 99.

3.1.3.2 Processes for mobile equipment and supplies

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.

There is no mobile equipment or supplies identified in connection with flooding.

3.1.3.3 Potential deviations from licensing basis

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

No deviations from the current licensing basis have been identified.

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.

As there is no fuel and therefore no heat generating source in the reactors and ponds no heat transfer is required.

3.2.2 Potential need to increase robustness of the plant against flooding

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

No need to increase the robustness of the plant against flooding has been identified.

4 Extreme weather conditions

4.1 Design basis

4.1.1 Reassessment of weather conditions used as design basis

4.1.1.1 Characteristics of design basis extreme weather conditions

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

The effects of extreme weather were assessed for the LTSR as Wind Loading and Extreme Ambient Temperatures. In each case, the assessment concluded that the hazard posed no threat to nuclear safety on the then operating Site.

The 2001 and 2011 PSRs also assessed both these extreme weather phenomena. It was concluded that with the removal of fuel, wind loading might cause some damage to the superstructure of buildings, but that time would be available to effect repairs before any potential existed for a release of radioactivity. Temperature extremes were of little concern since there were no temperature sensitive items of plant remaining. Moreover, time would be available to effect repairs before any significant threat existed.

The safety cases for the proposed Reactor Building Safestore (RBS), the new ILW store and the North and South Main Sludge Vaults all concede that extreme weather conditions could damage the external containment but judged that a radiological hazard was extremely unlikely to arise as a direct consequence. Prompt remedial action would be taken to mitigate and correct the effects of these hazards before any further degradation of protection occurred. In the case of the reactor buildings, although the vulnerability to wind damage is currently high pending the completion of repair work, the risk is still assessed as ALARP.

Flooding of the ILW facilities from internal faults is assessed in the relevant facility safety cases as water ingress faults (for instance rain water ingress). The only significant credible water ingress faults relate to the Fuel Element Debris (FED) vaults. Such ingress could lead to hydrogen evolution over time leading to a hydrogen explosion from an inadvertent ignition source. Such an explosion would not initiate a FED fire. Such a fault would require coincident failure of the ventilation system, the vault flooded for 20 days and the assumption that no hydrogen could leak from the vault during this time. The best estimate of the dose to the public would be 1.6µSv. The FED vaults have hydrogen monitoring alarms, ventilation systems and sumps. Actions in the event of increased hydrogen concentrations are covered in site abnormal operating instructions.

Consideration TRAW 5	Consideration will be given to the fire safety case for ILW storage facilities to identify any appropriate enhancements to the level of resilience.

The safety case for the defuelled Reactor Pressure Vessels (RPVs) concludes that the probability of RPV failure by brittle fracture is likely to be unaffected by periods of extreme low temperature by virtue of the high thermal inertia of the RPVs and the graphite cores. Furthermore the RPV fracture toughness is not greatly reduced by low temperatures. It should be noted that temperatures in the RPVs lag ambient temperatures by approximately three months.

4.1.1.2 Postulation of design basis characteristics

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

Climate change studies as reported in "Climate Change Scenarios for the United Kingdom", UKCIP02 Scientific Report, April 2002, ISBN 0 902170 60 0 predict higher temperatures and more frequent extreme weather events over the next 70 years. However, the increases are not judged to have a significant impact for the site. For the next PSR period of 10 to 15 years it is judged that the risks associated with these hazards remain ALARP.

It has recently been found that the Reactor Building structures do not meet the current wind code requirements for a 1 in 50 year ($2x10^{-2}$/y return frequency). Therefore it is possible that an extreme wind or a seismic event could lead to some part of the external walls collapsing on to adjacent buildings. The radiological consequence of a failure of the building structures has been shown to be above the broadly acceptable limit of risk to the public, but below the upper tolerable limit. A programme of remedial strengthening works has been instigated which will progressively reduce the risk. Local Safety Instructions are being reviewed to ensure that in periods of high wind the risk to workers is ALARP.

4.1.1.3 Assessment of frequency

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

As 4.1.1.2 above.

4.1.1.4 Potential combinations of weather conditions

Consideration of potential combination of weather conditions.

The simultaneous occurrence of more than one type of extreme weather of severity corresponding to 10^{-4} annual probability of exceedence has not been considered.

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.

As there is no fuel and therefore no heat generating source in the reactors or ponds, no heat transfer is required.

4.2.2 Potential need to increase robustness of the plant against extreme weather conditions

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

A programme of remedial strengthening work to ensure that the Reactor Building structures have sufficient capability to withstand an extreme wind event is being undertaken. It has been calculated that, as a consequence of building failure, a member of the public could receive a dose of 0.926 mSv resulting from liquid discharge and 0.182 mSv resulting from an airborne discharge. On completion of remedial strengthening the integrity of the buildings will be confirmed by periodic inspection in compliance with Site Licence Condition 28.

5 Loss of electrical power and loss of ultimate heat sink

5.1 Nuclear power reactors

For writing chapter 5, it is suggested that detailed systems information given in chapter 1.3. is used as reference and 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.

5.1.1 Loss of electrical power

5.1.1.1 Loss of off-site power

5.1.1.1.1 Design provisions taking into account this situation: back-up power sources provided, capacity and preparedness to take them in operation.

As there is no fuel in the Reactors electrical power is only required for monitoring.

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

As there is no fuel in the Reactors electrical power is only required for monitoring.

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

5.1.1.2.1 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.

As there is no fuel in the Reactors electrical power is only required for monitoring.

5.1.1.2.2 Battery capacity, duration and possibilities to recharge batteries.

As there is no fuel in the Reactors electrical power is only required for monitoring.

5.1.1.3 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

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

As there is no fuel in the Reactors electrical power is only required for monitoring.

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

As there is no fuel in the Reactors electrical power is only required for monitoring.

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

As there is no fuel in the Reactors electrical power is only required for monitoring.

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

As there is no fuel in the Reactors electrical power is only required for monitoring.

5.1.2 Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

As there is no fuel in the Reactors electrical power is only required for monitoring

5.1.3 Loss of the ultimate heat sink

5.1.3.1 Design provisions to prevent the loss of the primary ultimate heat sink

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.

As there is no fuel in the Reactors no ultimate heat risk is required.

5.1.3.2 Effects of loss of the primary ultimate heat sink

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).

5.1.3.2.1 Availability of an alternate heat sink

As there is no fuel in the Reactors no ultimate heat sink is required.

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

As there is no fuel in the Reactors no ultimate heat sink is required.

5.1.3.3 Loss of the primary ultimate heat sink and the alternate heat sink

5.1.3.3.1 External actions foreseen to prevent fuel degradation.

As there is no fuel in the Reactors no ultimate heat sink is required.

5.1.3.3.2 Time available to recover one of the lost heat sinks or to initiate external actions and to restore core cooling before fuel damage: consideration of situations with various time delays from reactor shut down to loss of normal reactor core cooling state (e.g., start of water loss from the primary circuit).

As there is no fuel in the Reactors no ultimate heat sink is required.

5.1.3.4 Loss of the primary ultimate heat sink, combined with station black out (i.e. loss of off-site power and ordinary on-site back-up power source).

5.1.3.4.1 Time of autonomy of the site before start of water loss from the primary circuit starts.

As there is no fuel in the Reactors no ultimate heat sink is required.

5.1.3.4.2 External actions foreseen to prevent fuel degradation.

As there is no fuel in the Reactors no ultimate heat sink is required.

5.1.4 Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

As there is no fuel in the Reactors no ultimate heat sink is required.

5.2 Spent fuel storage pools

Where relevant, equivalent information is provided for the spent fuel storage pools as explained in chapter 5.1 for nuclear power reactors.

5.2.1 Loss of electrical power

As there is no fuel in the storage ponds electrical power is only required for monitoring.

5.2.2 Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

As there is no fuel in the storage ponds electrical power is only required for monitoring.

5.2.3 Loss of the ultimate heat sink

As there is no fuel in the storage ponds no ultimate heat sink is required

5.2.4 Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

As there is no fuel in the storage ponds no ultimate heat sink is required.

6 Severe accident management

6.1 Organisation and arrangements of the licensee to manage accidents

Chapter 6.1 should cover organization and management measures for all type of accidents, starting from design basis accidents where the plant can be brought to safe shut down 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

6.1.1.1 Staffing and shift management in normal operation

As Trawsfynydd is currently undergoing decommissioning activities to prepare the site for entry into Care and Maintenance, the staffing levels are dependent on the projects being carried out at the time.

6.1.1.2 Plans for strengthening the site organisation for accident management

Arrangements have been produced for the management of emergencies on Trawsfynydd Site. The Trawsfynydd Emergency Plan is a document containing the site emergency arrangements and the arrangements for collaboration with external organisations, including the Regulators, the emergency services, local government and central government. The document is approved by the Regulator (ONR) and changes cannot be made to it without their formal agreement. The on-site arrangements allow for the establishment of an Emergency Control Centre (ECC) staffed by at least an Emergency Controller, an Emergency Administration Officer, an Emergency Health Physicist, an Emergency Technical Officer and an Emergency Communications Officer. In addition an Access Control Point and a Site Response Team are established. Members of these teams are on an emergency call-out rota using a telephone voicemail system based upon a stand alone off site server. Additional telecommunications can be made independently through BT or Mobile networks.

6.1.1.3 Measures taken to enable optimum intervention by personnel

The arrangements described above allow for the intervention of personnel to assess and mitigate emergency situations.

6.1.1.4 Use of off-site technical support for accident management

In the event of a site incident or off-site nuclear emergency being declared the Central Emergency Support Centre (CESC) is set up in Gloucestershire.

This dedicated facility is manned by a Controller, a Health Physicist and a Technical Officer each with a support team on a one-hour call out rota.

The remit of the CESC is to:

(i) Relieve the affected station of the responsibility for liaison with outside bodies on off-site issues in as short a time as possible after an accident.

(ii) Take over for the affected site at an early stage the task of directing the off-site monitoring teams and assessing their results.

(iii) Provide the requisite technical advice on off-site issues to all stakeholders in the Strategy Coordination Centre and those agencies represented in the CESC.

(iv) Provide regular authoritative company briefings for the media on all aspects of the emergency.

(v) Co-ordinate advice and support from within the affected company and other parts of the nuclear industry to the affected station.

(vi) Centrally manage the collation of all relevant information relating to the event (using appropriate means).

The CESC Controller has the full backing of the Company to take whatever steps are necessary, including using any resources required, to control the situation.

The Technical Support Team in the CESC has access to the Company Drawing Office so can obtain and print systems diagrams and a range of experts to help analyse the issues on-site and formulate recovery plans.

The CESC also has access to Procurement and the Supply Chain to obtain any goods or services required in the recovery.

The CESC manages the links to the local and national responding organisations.

The CESC takes over the management of the Off-site survey and the formulation of Company advice.

The CESC mobilises and coordinates the resources of the whole Company and cooperation from other nuclear companies.

6.1.1.5 Procedures, training and exercises

In accordance with arrangements agreed with the Regulator (ONR) training is given to site personnel with an involvement in emergency arrangements. Exercises are held to demonstrate the arrangements.

6.1.2 Possibility to use existing equipment

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

The Company shares a set of Beyond Design Basis Accident Containers, maintained at a central UK location, which can be transported to any affected site. These containers are equipped with Command and Control provisions and materials.

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

As there is limited need for the provision of fuel and other supplies there are no special arrangements for their management at the site.

6.1.2.3 Management of radioactive releases, provisions to limit them

As there is only a limited release from the stored radioactive waste, compared with releases from operational reactors, there are no special provisions.

6.1.2.4 Communication and information systems (internal and external).

In the event of an accident or natural disaster at a power station there is a need to be able to promulgate an alert and then to pass information into and out of the site. Particularly important communications paths are those between the site, the Strategic Coordination Centre (SCC), the Central Emergency Support Centre (CESC) and the responding emergency services.

The site's telephone system is designed to be resilient through the use of duplicated separate offsite connections. Telephones in the key response centre are not all connected to the same Public Switched Telephone Network (PSTN) exchange, thus ensuring that failure of an exchange would not leave the room without a working telephone. Also the site is within the coverage footprint of several mobile telephone networks, providing further communications diversity.

Equipment is provided for radiological monitoring (see section 2.1.3.2).

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

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

There is a single access route to the Trawsfynydd Site. Should this road become impassable there are other routes that could be utilised.

6.1.3.2 Loss of communication facilities / systems

The Company has robust communications systems featuring diversity and redundancy, particularly at operating sites. These include:

- A resilient Company Wide Area Network
- For operating sites – diverse routes to the outside world communications cloud.
- Telephones that are independent of the Company exchanges with direct (copper) links to the PSTN.
- The Nuclear Industry Airwave Service, designed to allow communication with off-site survey vehicles, can be used to make phone calls independent of the local PSTN.

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

In all exposure conditions including accident response, doses to personnel should be below dose limits (normally 20 mSv whole body dose) and must be As Low as Reasonably Practicable (ALARP). In the event of a major accident at a nuclear site the higher REPPIR Emergency Exposures can be applied to informed volunteers. The role of the Health Physicist in the ECC is to ensure the safety of all people on site.

Staff that are not responding to an accident will be subject to controls based on doserate, airborne contamination levels and other hazards, and may be evacuated from the site.

The ECC is positioned to minimise the likelihood that it would be damaged in an accident or affected by radiation. It would be subject to tenability checks, the Initial Control Dose limit being 10 mSv over the first 10 hours. After this period the situation would be reassessed in the light of the radiological conditions, availability of replacement staff, etc.

On-site survey and emergency team staff controlled from the Access Control Point (ACP) are subject to the normal dose limits but in the event of a major accident the higher REPPIR Emergency Exposures (whole body doses of 100 mSv for operations and 500 mSv for life saving) can be applied to informed volunteers. Health Physics monitoring provides information on the local dose rates allowing response teams to ensure their doses are minimised and Electronic Personal Dosimeters are used to monitor doses and enforce dose limits.

Training is given on the use of appropriate Personal Protective Equipment and undressing/decontamination processes, and use of these would not prevent appropriate remedial work being undertaken.

In some extreme instances high radiation levels could make access to the damage scene unachievable. If this were the case then remote access or the installation of the appropriate level of shielding would be required. If radiation levels remain high then working time would be limited, which could impair the recovery operation particularly if the operations required are time consuming. Under conditions of high local dose rates, contamination and destruction of some facilities the Company would be relying on the site Command and Control structures to manage the event making an accurate assessment of the situation and best use of available resource.

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

The site facilities used to manage emergencies are specifically designed for that purpose.

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

Key emergency response centres on site are the ECC and ACP.

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

The accident management measures provided at Magnox sites are intended to be flexible. Identified personnel have high levels of authority to utilise any resources available and technical advice is available from off-site facilities.

6.1.3.7 Unavailability of power supply

A description of power supplies is given in Section 1.3.

6.1.3.8 Potential failure of instrumentation

There is no permanently installed instrumentation required for emergency management.

6.1.3.9 Potential effects from the other neighbouring installations at site.

There are no other nuclear licensed sites operating in the area around Trawsfynydd site.

6.1.4 Measures which can be envisaged to enhance accident management capabilities

Additional measures have been identified for consideration.

Consideration TRAW 1	Consideration will be given to enhancing the availability of beyond design basis equipment
Consideration TRAW 2	Consideration will be given to providing further equipment to facilitate operator access around the Site
Consideration TRAW 3	Consideration will be given to enhancing on site arrangements for command, control and communications
Consideration TRAW 4	Consideration will be given to updating and enhancing severe accident management guidance

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

Not required as no fuel stored on site.

6.2.1 Elimination of fuel damage / meltdown in high pressure

6.2.1.1 Design provisions

Not required as no fuel stored on site.

6.2.1.2 Operational provisions

Not required as no fuel stored on site.

6.2.2 Management of hydrogen risks inside the containment

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

There is no fuel stored on site and no containment. It should be noted, however, that there are some very low probability events that could lead to the generation of hydrogen in one of the waste storage vaults. This is discussed in section 4.1.1.1.

6.2.2.2 Operational provisions

Operator actions are addressed in Plant Operating Instructions.

6.2.3 Prevention of overpressure of the containment

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

Not required as there is no containment.

6.2.3.2 Operational and organisational provisions

Not required as there is no containment.

6.2.4 Prevention of re-criticality

6.2.4.1 Design provisions

Not required as no fuel stored on site.

6.2.4.2 Operational provisions

Not required as no fuel stored on site.

6.2.5 Prevention of base-mat melt through

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

Not required as no fuel stored on site.

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

Not required as no fuel stored on site.

6.2.5.3 Cliff edge effects related to time delay between reactor shut down and core meltdown

Not required as no fuel stored on site.

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

6.2.6.1 Design provisions

Not required as there is no containment.

6.2.6.2 Operational provisions

Not required as there is no containment.

6.2.7 Measuring and control instrumentation needed for protecting containment integrity

Not required as there is no containment.

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

Not required as there is no containment.

6.3 Accident management measures to restrict the radioactive releases

6.3.1 Radioactive releases after loss of containment integrity

6.3.1.1 Design provisions

Not required as there is no containment.

6.3.1.2 Operational provisions

Not required as there is no containment.

6.3.2 Accident management after uncovering of the top of fuel in the fuel pool

6.3.2.1 Hydrogen management

Not required as no fuel stored on site.

6.3.2.2 Providing adequate shielding against radiation

Not required as no fuel stored on site.

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

Not required as no fuel stored on site.

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

Not required as no fuel stored on site.

6.3.2.5 Availability and habitability of the control room

Not required as no fuel stored on site.

6.3.3 Measures which can be envisaged to enhance capability to restrict radioactive releases

Not required as no fuel stored on site.

7 Glossary

ACP	Access Control Point
AETP	Active Effluent Treatment Plant
ALARP	As Low As Reasonably Practicable
CESC	Central Emergency Support Centre
DBE	Design Basis Earthquake
DBF	Design Basis Flood
ECC	Emergency Control Centre
ENSREG	European Nuclear Safety Regulators Group
FED	Fuel Element Debris
ILW	Intermediate Level Waste
LLW	Low Level Waste
LTSR	Long Term Safety Review
MAC	Miscellaneous Activated Components
MCP	Management Control Procedure
ONR	Office for Nuclear Regulation
PGA	Peak Ground Acceleration
PML	Principia Mechanica Limited
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSTN	Public Switched Telephone Network
RBS	Reactor Buildings Safestore
REPPIR	Radiation Emergency Preparedness and Public Information Regulations
RPV	Reactor Pressure Vessel
RV1	Resin Vault 1
SCC	Strategic Coordination Centre
SSC	Systems Structures and Components
UPS	Uninterruptible Power Supplies
UK	United Kingdom

Table 1 - List of Considerations

Consideration Reference	Item for consideration	Report Section
TRAW 1	Consideration will be given to enhancing the availability of beyond design basis equipment.	6.1.4
TRAW 2	Consideration will be given to providing further equipment to facilitate operator access around the Site.	6.1.4
TRAW 3	Consideration will be given to enhancing on site arrangements for command, control and communications.	6.1.4
TRAW 4	Consideration will be given to updating and enhancing severe accident management guidance.	6.1.4
TRAW 5	Consideration will be given to the fire safety case for ILW storage facilities to identify any appropriate enhancements to the level of resilience.	4.1.1