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An overview of the preparation and implementation of the safety case in support of the 2nd JET deuterium–tritium experiment

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Abstract

An overview of the preparation and implementation of the JET 2nd deuterium-tritium experiment (DTE2) safety case is presented. The safety case regime, developed by UKAEA for fusion applications to demonstrate compliance with relevant regulations, and its implementation on UKAEA sites is outlined. The hazard assessment process and details of the methodology applied to assess key JET fault scenarios are provided. An outline of key inventories, their composition, and applied factors for consequence and risk assessment are discussed. The consequences resulting from key fault sequences, the designation of appropriate safety measures and the associated risks are summarised and compared against basic safety limits and objectives. Finally, some of the lessons learned from DTE2 following implementation are discussed.

Keywords: safety case, tritium, assessment, methodology, loss of vacuum, loss of coolant, shielding

1. Introduction

A safety cases is a form of risk assessment which is produced to cover high hazard nuclear and bespoke/novel non-nuclear plant/process.

In August 2020 the JET device located at UKAEA Culham in Oxfordshire UK, implemented the 2nd deuterium–tritium experiment (DTE2) safety case [1]. The implementation of this safety case enabled full scale tritium operations to occur on JET for the first time since DTE1 in 1997 [2]. Following completion of DTE1, which was regarded as active commissioning, JET activities were undertaken under the 2001 Torus and Active Gas Handling System (AGHS) pre-operational safety report [3] which was succeeded by the Torus deuterium–deuterium (D–D) Safety Case [4] in 2011 to justify continued D–D operations following installation of the beryllium first wall.

Development of the Torus safety case for DTE2 operations commenced in 2011 with the production of a provisional safety case in 2014 [5, 6]. The provisional safety case assessed the feasibility of returning JET safely to tritium operations, identifying the primary candidate safety measures and associated actions required to demonstrate that the associated risks could be managed to as low as reasonably practicable (ALARP). Some specialist assessments and plant modifications were identified as a result of this case [4]. Changes to plant or process that impact the extant facility safety case are formally managed as safety case modifications. These are formally subsumed into subsequent issues of the facility safety case when a periodic safety review is carried out.

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[1] See the author list of ‘Overview of T and D-T results in JET with ITER-like wall’ by CF Maggi *et al* to be published in Nuclear Fusion Special Issue: Overview and Summary Papers from the 29th Fusion Energy Conference (London, UK, 16–21 October 2023).

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Following issue of the provisional safety case in 2014, the full DTE2 Torus facility safety case [1] was developed incorporating the design modifications to JET proposed to enable a return to full JET tritium operations. The DTE2 safety case reviewed operational experience from DTE1, and modifications to the plant since the 1997 campaign and utilised modern codes and standards to assess the risk associated with tritium operations for DTE2. The AGHS which is designed to supply, recover, process and recirculate the hydrogen isotopes, including tritium, which are used by the Torus, has its own separate facility safety case [7].

The preparation and implementation of the JET DTE2 safety case is described below in the context of the regulatory framework applied at UKAEA. Following the identification of hazards, there is a focus on the methodology applied for assessment of JET fusion hazards. Design safety measures are described, and key fault assessments are summarised. Fault consequences are discussed together with the impact of designated safety systems and controls. Associated doses and risks are discussed in the context of corresponding basic safety objectives and limits (BSOs and BSLs respectively). Finally, some of the key lessons learned from D–T operations with respect to the preparation and implementation of the DTE2 Safety Case are discussed.

2. Regulation and safety management

UKAEA operates a safety case regime to support its operational portfolio across the Culham and Rotherham sites. This provides a robust means to demonstrate that risks associated with high hazard operations are reduced against the ALARP principle. The safety case regime acts as UKAEA's risk assessment process for high hazard operations required under the Health And Safety At Work Act 1974 [8].

In addition to the Health And Safety At Work Act 1974, the safety case demonstrates compliance with the Ionising Radiation Regulations 2017 [9] which are implemented on the Culham site through Radiation Protection Rules and the Environmental Agency (EA) permit. The safety case hazard assessments provide input to the Radiation Emergency Preparedness and Public Information Regulations 2019 [10] report for the Culham site [11].

UKAEA operate an Authority To Operate (ATO) safety management structure. An ATO is issued to specifically appointed, suitably qualified and experienced individuals (called the ATO holder) who are responsible for ensuring that operations allowed by the ATO are conducted safely and in accordance with the relevant safety case limits and conditions.

An ATO is authorised by the UKAEA Chief Operating Officer following independent peer review of the relevant safety case documentation and advice from the Site Safety Working Party (SSWP). The membership of the Site Safety Working Party includes representatives from UKAEA departments responsible for safety quality and environment, operations, engineering, health physics (HP), and safety case engineering as well as members independent of the organisation.

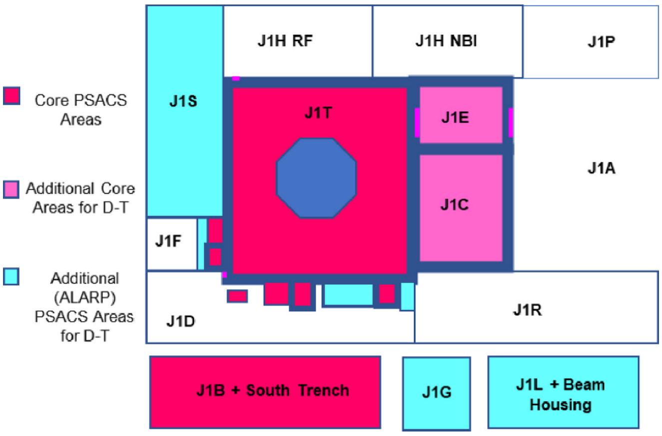

Figure 1. Layout of personnel safety and access control areas for JET.

2.1. Building layout

The JET Building complex was designed in accordance with relevant British Standard Codes of Practice, specifications and statutory requirements in force at the close of the design stage in 1978. As well as housing the plant and support systems in a suitable environment, the building is also required to protect plant and operators from hazards arising both inside and out. Figure 1 shows the arrangement for personnel safety and access control (PSACS) for the JET operational and shielded areas.

Personnel access to the JET Torus hall (J1T) and the JET basement (J1B) is prevented during JET pulsing operations due to the direct radiation hazard. It is necessary to also prevent personnel access to the JET entrance (J1E) and hot cells (J1C) during D–T pulsing operations when direct radiation hazards increase. Personnel access to the J1 support wing, the service tunnel (J1G) and the J1 roof laboratory (J1L) is also prevented during D–T pulsing operations due to localised direct radiation hazards and in the support wing (J1S), there is the potential for exposure to activated gases and permeated tritium associated with the baking plant.

2.2. Hazard identification

Hazard identification was drawn from a number of sources to provide a list of fault scenarios for JET. These included reviews of relevant accidents in fusion and fission reactors, HAZOP studies [12–14] and ongoing input from modification safety cases and reviews of incidents on the JET site. These have been collated into the 'Torus Fault Schedule'.

The key faults assessed relate to:

- Release of radioactive materials (tritium and mobile activation products);
- Exposure to direct and indirect radiation (neutron and gamma);
- Exposure to beryllium dust;
- Release of cryogens;

- Electrical hazards;
- Hydrogen deflagration.

Radiological faults have been assessed under both operations and shutdown plant regimes when different machine inventories and operator access arrangements apply.

The key fault scenarios are:

- Loss of vacuum accidents (LOVA) which considers gaseous release (with suspended dust) from the vessel;
- In-vessel loss of coolant accidents (LOCA) and loss of flow accidents which consider release of coolant with potential to lead to a release from the vessel;
- Ex-vessel LOCA which consider release of activated coolant;
- Plasma heating and fuelling systems events (PHFSE) which may lead to a release of tritium and activated gases as well as combustible gases such as deuterium;
- Magnet events i.e. faults which may lead to releases of radioactive material through melting of coils/insulation;
- Shielding events i.e. faults where operators may be exposed to high neutron and gamma fluxes during a pulse and may be exposed to gamma fluxes from activated materials following pulsing;
- Loss of plasma control (LOPC) resulting in disruptions which can lead to failure of the vacuum boundary;
- Loss of electrical power leading to system failures which may lead to release of radioactive material;
- External events such as earthquakes, aircraft crashes, missiles, extreme weather etc.

In addition, there is an assessment of fault scenarios associated with operator maintenance, intervention and shutdown tasks as well as a normal operations assessment.

The AGHS is linked to the Torus by a pipe bridge which carries the matrix lines for recovery of Torus exhaust gases, as well as the supply lines to fuel the Torus. Following processing and recovery of tritium within the AGHS, the AGHS exhaust detritiation system (EDS) provides abatement of residual discharges to the environment, and emergency pumping and detritiation for the Torus under fault conditions. The boundary between Torus and AGHS ATOs is defined within an interface definition local rule which identifies specific AGHS supply and Torus exhaust isolation valves. Interfacing faults associated with supply and recovery of tritium are assessed within each facility safety case with respect to consequences within the separate ATO areas.

3. Hazard assessment methodology

3.1. Inventory at risk

In addition to the maximum inventory that can be stored on the cryopanels during operations (11 g) which is readily mobile, some of the tritium held-up in machine surfaces could be released on exposure to moist air or coolant as a result of a LOVA or LOCA. By assessing tritium hold-up during DTE1 operations and tritium recovery following subsequent clean-up, and vent, soak and purge [15–17], a conservative value of potentially releasable tritium (4 g) from hold-up was derived. A maximum tritium inventory of 15 g was therefore applied in design basis assessments (DBA). A maximum hold-up of 11.5 g tritium (including a residual 1 g tritium from DTE1) was predicted for DTE2 in the safety case [1], of which 35% (4 g) was assumed releasable in the event of a LOVA/ LOCA (based upon tritium recovery during vent and purge following DTE1). This was then scaled to DTE2 throughput (10 times higher) with an assumed reduction in retention for the DTE2 beryllium first wall in comparison to the carbon first wall in place for DTE1 (10 times lower), justified by fuel retention gas balance experiments and post-mortem analysis of in-vessel components removed in shutdowns [18, 19]. Since this inventory will cycle between 4 and 15 g, an average releasable inventory of 10 g was applied for probabilistic risk assessments (PRA). This is summarised in table 1.

Shutdown faults in the DTE2 safety case apply conservative tritium in air (3 $\mathrm{GBqm^{-3}}$) and a dose rate of (1 $\mathrm{mSv\ h^{-1}}$) set as radiological constraints for 'emergency' JET vessel access for DTE1. For DTE2, these assumed comparable clean-up and elapsed time as for DTE1. Tritium in air levels assumed EDS provides a minimum of 1 air change per hour (acph) for the vessel. A mid-plane direct radiation dose rate of 0.35 $\mathrm{mSv\ h^{-1}}$ is applied as a dose rate constraint for planned JET vessel access.

Historically a maximum releasable dust inventory of 1 g per Kg deposited has been assumed for JET LOVA/ LOCAs in the safety case [3]. Specifically, this was based on analysis of JET shutdowns and measurement of suspended dust levels on venting the machine reported in 1994 [20, 21]. If all deposited films are considered as potentially dust i.e. with low mobilisation factors, since on average 0.5–1 Kg of deposited film has accumulated in the divertor per campaign [22, 23], it was estimated that there could be between 2–5 Kg deposited by DTE2 [24]. A 5 g mobilizable dust inventory is therefore conservatively applied in the safety case for DTE2. Specific dust activities were calculated for ^{3}H, ^{49}V, ^{54}Mn, ^{47}Co, ^{58}Co, ^{60}Co, ^{59}Ni and ^{63}Ni. A specific dust activity of 2.0×10^{7} $\mathrm{Bqg^{-1}}$ is applied for 100% ^{60}Co in Safety Case DBA assessments. This applied the highest specific activity for a single isotope (2.0×10^{7} $\mathrm{Bqg^{-1}}$ for ^{58}Co) to ^{60}Co (for which inhalation and ingestion dose coefficients are a few orders of magnitude greater than the other radionuclides identified) and assumed 5.0×10^{23} neutrons per year [25, 26]. This approach was more conservative than summing doses for individual radionuclides and incorporating short lived isotopes. For PRA assessments this was reduced to 2.0×10^{5} $\mathrm{Bqg^{-1}}$ since DTE2 neutron yield would not exceed 2.0×10^{21} neutrons. Whilst it was anticipated that less deposit would be introduced following replacement of the CFC tiles with beryllium tiles, this could be off-set by e.g. higher power pulses. In addition, residual CFC dusts have accumulated beneath the divertor. Application of CFC assumptions was therefore considered appropriately conservative for safety case assessments.

A summary of the main inventories applied for operational faults is presented in table 2. Inventories are also derived for

Table 1. Summary of initial safety case predictions for tritium hold-up and release for DTE2.

		Mass T (g)	
Scenario	Comment	DTE1 measurements	DTE2 predictions
DTE1 measurements			
T held-up at end of campaign	30% of throughput injected into plasma	10.5	
T retained after clean-up pulsing	17% of throughput injected into plasma	6.0	
T deemed released on vent	33% of tritium retained after clean-up	2.0	
T remaining before DTE2	Conservative assumption not taking into account decay, tile removal and tritium in condensate.	1.0	
DTE2 predictions based on DTE1 with 10% retention and 10 × 35 g injected directly into plasma*			
Additional T held-up at end of DTE2 campaign	3% of throughput injected into plasma		10.5
Total T held-up at end of DTE2	DTE2 + DTE1		11.5
Conservative assumption for release from hold-up in event of LOVA during DTE2	Conservatively taken as 35% total hold-up		4.0
Inventory at risk in LOVA during DTE2	Release from wall plus cryopanel		15.0

Table 2. Summary of the main inventories applied for operational faults.

Inventory	Releasable Inventory during operations/g	Characteristics	Comment
Tritium	DBA: 15 g PRA: 10 g-	360 TBqg^{-1}	Release compositions range from 99:1% HT:HTO to 100% HTO. EDS detritiation assumes 95% conversion of HT to HTO and a detritiation factor of 1000 for HTO
Tokamak dust	5 g	DBA: 2.0×10^7 Bqg^{-1} PRA: 2.0×10^7 Bqg^{-1}	1 g per kg accumulated assumed to be released. HEPA filtration not credited in dose assessment

tritiated coolants, baking loop gases, activated Galden coolant, getter dust and activated electrical insulator gases (SF_6 and NOVEC).

3.2. Assessment process

Both design basis (DBA) Accident analysis and PRA is carried out for JET. DBA is used to assess unmitigated doses for categorisation and designation of safety systems and controls. Safety related equipment (SRE) and safety management requirements (SMRs) are designated in alignment with BSLs (20 and 1 mSv for workers and members of the public respectively) and BSOs for normal operations doses (1 and 0.01 mSv for workers and members of the public respectively) [27]. In this way DBA assessment is used to demonstrate that for single failure events with the identified designated controls in place, both on and off-site doses are reduced to well below the BSLs (requires key safety systems or controls (KSRE/ KSMR)) and below the BSOs (requires SRE or SMR). Consideration of event frequency and unmitigated consequence enables comparison against on and off-site DBA Regions for demonstration of sufficiency of safety measures for individual fault scenarios. For JET BSOs and BSLs are applied to summed frequencies of off-site doses in 0.01–0.1, 0.1–1, 1–10 and 10–100 mSv dose bands.

PRA enables the calculation of the nominal sum of risks to operators and the individual risk to members of the public which can be compared against nuclear accident risk targets (BSOs) and BSLs [27] summarised in table 3. For the JET Facility, a BSO and BSL of 1.0×10^{-06} y^{-1} and 1.0×10^{-04} y^{-1} applies for both on and off-site risks. PRA is applied in line with UKAEA procedures for assessment of nuclear fission facilities. PRA doses are used to calculate risks to operators and members of the public for each individual fault sequence, including dual failure of safety measures. The risks are then summed.

Demonstration that the risks to members of the work force and public are ALARP also requires demonstration of safety measures to minimise dose and discharges. These include key safety features inherent in the design, safety management arrangements for operations and maintenance, fault detection and recovery processes, monitoring and control systems as well as the designated safety systems and controls.

3.3. Dose assessment

Operator doses consider inhalation on exposure to airborne contamination, blotter and diffusion intake on exposure to tritiated water spills, wound transfer, ingestion and direct

Table 3. Summary of dose coefficients used in the hazard assessments.

Dose coefficient/Factor	Value (SvBq^{-1})	Application
Inhalation committed dose coefficient (HT) Inhalation committed dose coefficient (HTO) Inhalation committed dose coefficient (^{60}Co)	1.8×10^{-15} 1.8×10^{-11} 1.7×10^{-08}	Operator, OSW and members of the Public DBA and PRA doses. DBA doses used for designation of (K)SRE/(K)SMR where worst-case (high HTO composition). PRA doses used for risk. PRA dose to members of the public used for summed frequency of doses on comparison against PRA criteria.
Wound committed dose factor (HTO)	1.8×10^{-11}	Operator DBA and PRA doses
P42 societal consequence dose coefficient (HT)	2.5×10^{-19}	T8 dose to members of the public. T8 dose to members of the public used for summed frequency of doses on comparison against PRA criteria. Applied for designation of (K)SRE and (K)SMR

radiation. Best practice committed effective dose coefficients published by ICRP are applied [28–30]. Best practice values of external dose rate conversion factors for different nuclides are applied for assessment of direct radiation [31, 32]. For members of the public both DBA and T8 [ONR SAP Target 8] doses are calculated[2]. The latter calculate the societal consequence i.e. lifetime (50 years) committed dose to an adult person situated 1 km downwind of the release, from all pathways with no countermeasures. These are derived for all applicable tritium release forms and take into account atmospheric conversion and are more conservative for low HTO tritium compositions. PRA dose calculations apply the same committed effective dose coefficients as applied for DBA but with different dispersion coefficients and release fractions as appropriate (see sections 3.5 and 3.6).

3.4. Tritium composition

The radiological impact of releasing tritium gas to atmosphere depends upon the rate of formation of tritiated water vapour (HTO) which is significantly more radiotoxic to humans than tritium gas (HT) and gives rise to much higher inhalation doses. On the basis of two references which indicate conversion rates for HT to HTO of less than 0.2% for tritium on cold surfaces in the presence of steam and for tritium on heated surfaces [33, 34], together with a further reference which [35] concluded that a conversion rate of 1% with conservative meteorological data would provide an upper bound for off-site doses, the composition of a tritium release from the Torus machine when at temperature is assumed to be 99:1% HT:HTO for e.g. a LOVA pressure driven initial release. An increase to 90%HT:10%HTO is assumed where there is a slow diffusion from the hot Vessel and release into the Torus hall prior to external release. For a LOVA during pulsing, it is assumed that tritium held up in the walls would be 100% converted to HTO prior to release, as would tritium released in a fire/explosion scenario.

The shutdown safety assessments [1] assume 100% HTO for releases as there may have been exposure to moist air for extended periods. Assessment of operator doses on release from interspaces assume a 90%HT:10%HTO as exposure to moist air will be instantaneous on breach.

For releases that are recovered through the EDS, it is conservatively assumed that the recombiner is 95% efficient for conversion of HT to HTO, and that a detritiation factor of 1000 is achieved [7]. For detritiation failure, a 10% conversion efficiency is conservatively assumed with no detritiation.

3.5. Dispersion coefficients

UKAEA derived dispersion coefficients for releases into operational areas which take into account the amount of mixing due to recirculating ventilation and the amount of isolation due to extraction [1]. When ventilation rates are taken into account, in some circumstances the total exposure time could exceed the time taken for the release cloud to fill the room. In such cases one dispersion coefficient is calculated applying the cloud expansion equation [36–38] for the time taken for the release to expand to the free space in the room, and a second dose coefficient is calculated for the subsequent dilution of the release due to extraction, applying an exponential decay equation. An overall dispersion coefficient is then derived by adding the two component dispersion coefficients together. In this way, dispersion coefficient for JET areas were derived for operations and shutdown ventilation scenarios. PRA dispersion coefficients assume reduced exposure times and increased exposure distances for operators.

Outside building dispersion coefficients are applied for on-site workers (OSWs) and members of the public. These are as originally calculated, using the Gaussian Plume dispersion model as recommended by the UK Atmospheric Dispersion Modelling Working Group [39]. They take into account the release point, the relative heights of the buildings and stacks, and the exposure distance. Releases recovered via the EDS would be discharged from the AGHS building

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[2] UKAEA derived P42 dose conversion factors based on 1992 SAPs Principle 42 (superseded as ONR SAPs Target 8) for nuclides on its sites. For example, dose conversion factors of 2.5E-19, 4.7E-19 and 2.4E-18 Sv/Bq are applied for elemental tritium, tritiated water vapour and insoluble particulate releases in the DTE2 and AGHS safety case hazard assessments.

stack which is treated as an isolated stack, whilst releases into the Torus hall would be discharged from the Torus building monitored stack which is treated as a building release, taking into account relative building to stack heights. Maximum single weather i.e. worst case downwind weather conditions are applied for DBA dispersion coefficients and mean downwind weather conditions are applied for PRA dispersion coefficients. Exposure distances of 200 and 250 m (reflecting the position of the site boundary fence, i.e. the nearest point that the public might reasonably have access to) are conservatively applied for DBA and PRA dispersion coefficients respectively. T8 doses consider exposure at 1 km downwind of the release.

3.6. Release fractions

For operator tasks, conservative release fractions of 1 for gases and 0.01 for liquids are applied for DBA with less conservative dose coefficients derived for dust and liquid spill PRA scenarios. JET LOVA and LOCA release fractions are derived from MELCOR modelling simulations using a fusion specific version of the MELCOR severe accident code version 1.8.5 adapted for application for JET [40, 41]. This modelling code has been widely used for thermal hydraulic analysis of severe accident analyses of commercial nuclear fission power plants.

3.7. Direct radiation

Direct radiation doses for operational shielding faults are scaled from original JET assessments [26] conservatively based on a maximum campaign of 5.0×10^{23} neutrons per year generated at 14.1 MeV for D–T pulsing. The maximum available neutron budget for DTE2 was conservatively assumed to be 1.7×10^{21} neutrons per year, with a maximum D–T pulse of 1.0×10^{20} neutrons [1]. Some more recent MCNP [42] and FISPACT [43] neutronics assessments were carried out prior to DTE2 for shielding assessments and shutdown dose assessments [1, 44, 45] and [46]. For fault scenarios, exposure to a maximum single pulse or a day exposure of 19 average pulses (5×10^{18} neutrons) and one maximum pulse is applied [1] depending upon the fault scenario.

3.8. Risk factors

NRPB risk factors [47] are applied for workers and members of the public. Conservative wind factors (0.1 and 0.25) are applied in the calculation of risk to members of the public and OSWs respectively. Occupancy factors are applied in the calculation of risk to operators.

4. Design safety measures

JET pulses generate large fluxes of high energy neutrons and gammas. Design safety measures to protect operators from direct radiation include the shielding elements that comprise the bulk radiological shield around the Torus hall, shielding doors and beams and removable elements which shield penetrations. PSACS prevents exposure to JET pulsing radiation by ensuring a pulse cannot be initiated until all shielding elements (doors, beams and blocks) are in place. This system prevents access to PSACS controlled areas during pulsing. PSACS is also interlocked with other high hazards such as HV and lasers. There is a pre-pulse warning siren and warning beacons and there are emergency stop push buttons placed in operational areas to prevent a pulse. Radiological protection instrumentation (RPI) is strategically placed to trigger evacuation via local and remote hardwired alarms when excess levels of gamma radiation are detected.

The primary role of containments used at JET is that of controlling tritium release to the environment. The design philosophy is that where practicable, double walled, all-metal construction is used for tritium containment. There is provision of high integrity primary and secondary containment for tritium throughout JET and peripheral plants. The JET vacuum vessel is a double-walled Nicrofer 7216LC alloy structure designed for ultra-high vacuum conditions (10–9 mbar) required to achieve plasma discharges. The Torus machine itself can be regarded as primary containment. Forced ventilation in the Torus hall and basement maintains a negative pressure gradient. This and the efficiency of the seals between the shielding structures are designed to contain releases within the Torus hall/basement such that any release would be discharged through the building monitored stacks. In the event of overpressure within the Torus vacuum boundary, there is a pressure interlock cascade designed to ensure that leaks are isolated and directed to the EDS rather than to Torus operational areas. This system provides hardwired pressure relief and detritiation of discharges. The drain and refill system (D&RS) enables leak detection and isolation. An automated hardwired LOVA alarm has been installed to initiate evacuation of operational areas in the event of overpressure, prior to a potential tritium release. Finally, RPI is strategically placed to trigger evacuation via local and remote hardwired alarms when excess levels of tritium are detected.

5. Key fault sequences and designated safety systems and controls

Significant faults are those faults which, if unmitigated, could result in doses in excess of the Office for Nuclear Regulation (ONR) BSLs. Only three significant operational fault sequences were identified in the Torus D–D safety case [4]; pulsing with the shield doors open, an operator trapped within the Torus hall during pulsing, and a hydrogen deflagration event. For tritium operations, the list of such faults is extended to include LOVA and LOCA releases, faults associated with the gas introduction system (GIS), and some additional loss of shielding faults.

5.1. Shielding faults

JET shielding was designed to restrict the dose rate at the site boundary to less than 5 μSv y^{-1} and the dose rate for an individual penetration to less than 0.5 μSv y^{-1}. For operational areas the shielding was designed to reduce dose rates to less than 1 mSv y^{-1} to enable ‘free’ access, and to less than

66 mSv y^{-1} to enable controlled access (to a HP controlled area). Neutronics assessments were carried out to achieve this on the basis of the proposed neutron budget and planned operations for JET campaigns [42, 48]. [The shield wall made up of 2.5 m concrete lined with 300 mm of borated concrete was assessed to give dose rates of 0.23 mSv y^{-1} for a campaign producing 5×10^{23} neutrons y^{-1} [48]].

D–T pulsing with the doors open and exposure of an operator in the Torus hall during D–T pulsing have the potential to deliver the highest doses to an operator out of all the fault sequences analysed. A maximum DBA dose of 7.2 Gy is calculated for an operator in the Torus hall exposed to a single pulse. However, following pre-operational checks on shielding, PSACS mechanical interlocks prohibit pulsing if the shield elements are not secured. PSACS areas are confirmed as free from personnel by the pre-operational interactive search procedure, backed up by the DOSACS turnstile system. Should the search fail, PSACS warning beacons would alert any remaining operator to use the emergency stop push buttons and exit the area. Given these controls which are designated as key safety measures (KSRE and KSMR), the probability of these accidents occurring is very low and associated risks are below the BSO.

5.2. *LOVA*

On breach of the Torus vacuum boundary and spontaneous loss of vacuum, there is the potential for release of inventory into operational areas should safety controls fail.

A spontaneous breach could occur as a result of a failure of a window, bellows or feed through. Following air ingress and expansion within the hot vessel, there is potential for release of inventory back through the breach. The pressure interlock cascade (KSRE) would be triggered in response to the unplanned pressure rise to bypass and isolate the turbopumps (TMPs) and open up the exhaust route into the matrix lines and direct the vessel inventory to the EDS (KSRE). Response procedures direct operators to close the Torus isolation valves of individual diagnostic systems via a ‘global inhibit’ switch and to close the rotary high vacuum valves (RHVVs) to isolate the Torus from the neutral injection beams (NIBs). This would also isolate beamlines from the Torus. EDS emergency pumping is then automatically initiated once the EDS inlet pressure is reached (approximately 880 mbar(a)). Inhalation doses to both operators and members of the public could exceed the BSLs if key controls fail, preventing EDS pumping or detritiation.

MELCOR modelling [49] was used to assess air influx, the resultant pressure transients taking into account the automatic safety systems and operator response actions, to determine the rate and volume of any efflux back through the breach. This updated the hand calculations applied for the DTE1 SC. A range of breach sizes, temperatures and isolations, and dual failures of controls and response actions were assessed. The model uses a multi-compartmental approach to reflect the JET system and simulations were run until the system reached thermal convergence. The output enabled estimation of both DBA and PRA release fractions taking into account exposure times for on-site doses and considering releases at thermal convergence and beyond for off-site doses. Influx rates were compared with isolation rates to determine pressure transients and potential for efflux from isolatable systems to facilitate dual failure analysis. The MELCOR analysis revealed only one single failure scenario for which EDS pumping capability would not prevent a release into an operational area. This was for a breach greater than 100 mm diameter with both RHVVs closed at the maximum potential operating temperature of 320 °C. In practice normal operations are conducted at 200 °C–220 °C and access to the Torus hall is prevented at temperatures above 250 °C in a tritium regime. A key control (KSMR) requires regeneration of the cryopanels prior to access to the Torus hall (outside pulsing operations) to reduce the inventory at risk. This, and detritiation of inventory directed to EDS (KSRE) ensure that discharges as a result of a single failure are reduced to below the on and off-site BSOs.

An efflux of between 15%–25% of the inventory depending on the breach size is conservatively assumed for dual failure events that prevent EDS emergency pumping. The key controls discussed for shielding events prevent operator access during pulsing when oxidation of tritium held up in the vessel walls would increase inhalation doses. The requirement to regenerate Torus and NIB cryopanels (and recover gases to AGHS) prior to access (KSMR) then reduces consequent operator doses outside pulsing for the dual failure events to below the BSLs although BSOs would be exceeded (operator doses up to 10 mSv could result). Cryopanel flammable gas inventory limits (KSMR) are set to prevent a deflagration in the event of a LOVA when the resultant conversion to 100% HTO would increase inhalation doses significantly. However, tritium inventory limits (KSMR) are also set for tritium operations such that multiple failures would be required for a LOVA deflagration to occur.

Worst case doses to members of the public and OSWs occur in the event of a LOVA during pulsing. With all the safety controls operating successfully, single failure events could result in a T8 dose to a member of the public (in the order of 50–84 μSv), exceeding the BSO for off-site doses but well below the BSL. Corresponding doses to OSWs are below the BSO for onsite doses (a maximum of 0.5 mSv could result), with the exception of a worker on the roof, where an inhalation dose of 1.5 mSv is feasible. However, access to the roof is prohibited during pulsing and controlled by PSACS (KSRE) during tritium operations. Dual failure of designated safety controls (on failure to direct inventory to EDS or failure of EDS pumping or detritiation capability) could result in doses to member of the public and OSWs in excess of BSLs (up to 1.8 mSv and 8 mSv respectively).

A review of JET windows, bellows, feedthroughs and beamlines was carried out to refine IE frequencies (derived by summing their failure frequencies). The review also confirmed isolation of diagnostics that are not tritium compatible, and confirmed that windows where failure would result in a breach of the Torus vessel greater than 3.7 mm diameter had double containment [for LOVA of 3.7 mm diameter or less, pressure relief is not required]. On the basis of safety case PRA assessments, prescribed values were set for window failures during operations. Apart from spontaneous failures of vulnerable components, LOVAs could result from a disruption event (LOPC) which are assessed separately. In practice, these are likely to result in very small breaches, given the mitigation in place, and the associated risks are assessed to be lower.

Following a TMP rotor burst causing catastrophic failure of a TMP bellows prior to DTE2, the safety case introduced additional controls for the TMPs, identifying protective bellows liners, magnetic shielding and torque restrictions designated as SRE, with corresponding checks on the positioning of shields and restraints prior to operations being designated as SMRs. Neutron shields were also installed to protect the pump seals.

Development of the safety case LOVA analysis prompted revised flammable inventory controls, extension of the core PSACS areas to include additional core areas and ALARP areas during tritium operations, temperature restrictions for access and a requirement to regenerate the cryopanels prior to access. The local LOVA alarm was introduced (KSRE) to alarm in response to an unplanned pressure rise to initiate evacuation prior to a tritium release such that reliance on RPI monitoring (SRE) is reduced.

As confirmed by the PRA, dual or multiple failures are required to result in significant doses such that risks to members of the public and workers are well below the corresponding BSOs.

5.3. *LOCA*

A range of events could result in a loss of coolant into the Torus vessel or NIB box, with the potential for loss of containment and release of radiological inventory. The assessments take into account the very different Torus and NIB conditions and consider water and liquid helium/ nitrogen coolants as well as potential SF_6 and pressurised air discharges. As for the LOVA assessment, MELCOR modelling has been used to update the analysis [50, 51] and [52]. Creation of the models led to a clearer understanding of cooling circuits and draining systems, interfacing geometries, operational issues, and sequencing of protection/safety systems. Conflicting design intents were revealed for some machine protection and safety systems. This generally related to prevention of freeze-up in the event of small operational leaks (below or bordering on the threshold of scenarios considered in the safety case). For the LOCA models, the actions of a number of safety systems impact the fault sequences. The pressure interlock cascade (KSRE) provides pressure relief via EDS (KSRE) and triggers isolation of coolants. The vessel water coolant circuit D&RS and direct plant interlock system enable leak detection and isolation (KSRE). The NIB water cooling system has a water PLC and a DPIS to carry out isolation and leak detection with precautions to protect against freeze-up. Both cryogens and water supplies undergo initial isolation in response to an unplanned pressure rise.

No single failure LOCA gives rise to a dose to a worker above the BSO (a maximum dose of 22 μSv to an OSW could result). The only single failure LOCAs that result in a dose to members of the public are cryogenic LOCAs in the vessel or NIB, with the potential to result in a maximum T8 dose of 0.52 mSv if cryogenic swamping of the EDS is conservatively assumed for NIB cryogen LOCAs[3]. Dual failure of designated safety controls (on failure to isolate coolant supplies, failure to relieve pressure via EDS or failure of EDS pumping or detritiation capability) could result in DBA doses to operators or T8 doses to member of the public in excess of BSLs (doses up to 61 mSv and 1.5 mSv respectively could result). A LOCA induced LOVA deflagration event with 100% HTO release could result in higher doses, however multiple failure of controls would be required.

On installation of the beryllium wall, the potential hydrogen production as a result of beryllium-steam reaction was assessed [53]. It was concluded that the very rapid cooling of the hottest tiles in the vessel following a pulse, combined with the limited initial steam pressure at the tile surface made it unlikely that a self-sustaining Be-steam reaction could be established.

Both the nominal sum of risks to operators and the individual risk to a member of the public due to LOCA events are well below the BSOs. The LOCA assessments introduced new safety controls for additional coolant isolations and operator response actions, and introduced some additional commissioning checks.

5.4. *GIS faults*

Modifications for DTE2 introduced new tritium injection modules (TIMs), and additional tritium reservoirs and lines within the Torus hall and basement, increasing the tritium inventories present as free gas. A release from this system could result in a DBA dose to an exposed operator or members of the public in excess of BSLs or BSOs respectively. Higher doses could occur, significantly exceeding BSLs, in a fire if containment were lost.

The GIS has both primary and secondary containment (KSRE), with a nitrogen gas purge in the secondary containment which would direct inventory to a buffer tank in AGHS (AGHS SRE). A hardwired tritium alarm (KSRE) would alert operators to a release into secondary containment and area tritium monitors (SRE) would alarm on dual failure of secondary containment and release into an operational area. A dual failure is therefore required for a leak to result in a dose to operators. Since a new control was introduced requiring tritium to be present in the GIS to be withdrawn to the AGHS prior to Torus hall access (KSMR), in practice, multiple failures would be required. The JET building has a very early smoke detection system which will ensure that operators would be prompted to evacuate in the event of a fire prior to a tritium release, and procedures to minimise combustibles (KSMR). A specialist fire risk assessment was carried out [54] which concluded that any initiating fire event would burn itself out before reaching a sufficient intensity to compromise the integrity of the TIM or transferline containments since an impinging fire would need to be sustained for almost an hour. Actions to isolate feedlines (and reservoirs) to and from the gas distribution box (KSMR)

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[3] With no credit for EDS inlet air feed capability.

and actions to shutdown the ventilation, close dampers and cut power would mitigate the consequences. In addition, a nitrogen fire suppression system places the Torus Hall under a nitrogen atmosphere during tritium pulsing operations such that initiation of a fire would be unlikely. A new requirement for a pressure rise test of the GIS secondary containment prior to filling the primary transfer lines (interfacing SMR) was introduced to prove the integrity of the secondary containment prior to operations.

Faults associated with other plasma heating and fuelling systems were found to result in LOVA and LOCA events. Both the nominal sum of risks to operators and the individual risk to a member of the public due to PHFSE events are well below the BSOs.

5.5. *Operator tasks faults*

An assessment of operator tasks carried out during operations (outside pulsing operations) and shutdowns is presented in the DTE2 safety case. While individual tasks are carried out under HP advisement, some generic task assessments were performed. Tasks performed when the machine is under vacuum and at temperature could result in a LOVA in the event of a dropped load onto the vessel from the 150 ton crane or a load collision. However, the KSMR requiring regeneration of the cryopanels prior to access to the Torus hall (and hence prior to lifting operations) means that a significant release would require dual failure of designated controls and would result in a dose below the BSL (a DBA dose to an operator of 10 mSv or T8 dose to a member of the public of 0.39 mSv could result). Similarly the requirement to evacuate the GIS transfer lines and reservoirs prior to access (KSMR) prevents a significant dose in the event of dropped load on the GIS. An unplanned vessel breach (such as inadvertent valve opening) leads to similar consequences although EDS pumping (KSRE) can be claimed such that most of the inventory would be exhausted through EDS.

During shutdown periods, no single failures lead to doses to members of the public above the BSL (a maximum dose of 0.02 mSv applies for a release of permeated tritium from a diagnostic interspace). This is primarily because the release inventory is reduced (spread of contamination is minimised by removal of dust prior to entry and vessel pump and purge (KSMR) is carried out to minimise off-gas tritium), and also because containment is achieved by establishing a differential pressure gradient that ensures that leaks will be inward. In addition, the vessel exhaust will be detritiated in the EDS (KSRE) prior to discharge.

Administrative controls limiting tritium vessel/ interspace inventories prior to breach or vessel access (KSMR), appropriate RPE including pressurised suits (KSRE), and the use of Enclosures ensure that doses to operators are minimised. Single failure events resulting in doses above the BSL require extreme and unlikely failures such as failure to wear a pressurised suit/suit failure, failure of air supply during in-vessel access (KSRE) or falling through an enclosure roof. The EDS pumping and detritiation capability (KSRE) and machine containment (KSRE) limit both on and off-site exposure to contamination and reduce release inventories. The Enclosure ventilation systems, port and enclosure seals, and Torus hall area ventilation (SRE) ensure a differential pressure gradient to minimise spread of contamination away from the vessel. Stack tritium monitors (KSRE) would alert operators to a tritium release so that evacuation of affected areas can be initiated.

The nominal sum of risks to operators for the facility is dominated by the risks arising due to operator tasks which for tasks carried out under a shutdown regime exceed the BSO but are well within the BSL. The risks during a shutdown require consideration of the operator pool carrying out individual tasks since more specialist work pools apply. In practice, planning of shutdown tasks and application of dose constraints ensures that the risk to the most exposed worker is managed to be within the BSO. The nominal sum of risks due to operator tasks during operations is well below the BSO.

6. Implementation of the safety case

Key SMRs and SMRs ((K)SMRs) are implemented primarily through JET local rules, operating instructions (OIs), and machine operations group documentation emergency response and commissioning procedures which were all updated for DTE2. The wording of the (K)SMRs was agreed following discussions with operators, the ATO holder, safety case team and the peer reviewer to ensure that they are practicable, clear and effective. Prior to DTE2, operator tasks were reviewed by an human factors specialist [55] and (K)SMRs impacting day to day operations (primarily associated with pre-operational search, pre-operational checks, staffing and access arrangements) were trialled in the D–T technical rehearsal [56]. The D–T rehearsal also provided an opportunity to trial ventilation arrangements and tasks associated with operation of the nitrogen fire suppression system for the Torus hall which was reinstated for DTE2.

Implementation of Key SRE and SRE ((K)SRE) is managed through an examination, maintenance, testing and inspection (EMIT) schedule, with many of the key controls being tested in commissioning procedures prior to operations. Fitness for purpose assessments of (K)SRE were initiated in the provisional JET D-T safety case [5]. These established capability to perform the designated safety function, ongoing maintenance requirements, and identified any shortfalls and improvements.

Readiness for tritium operations was managed through a hold point control plan. A readiness review was carried out by the AGHS chief engineer for each of the AGHS sub-systems culminating in a hold point for AGHS operations with full reinstatement of the new EDS. Completion of a number of DTE2 safety case implementation Actions was also required for expansion of the Torus tritium boundary which included revision of all relevant rules, instructions and procedures and associated operator training, completion of outstanding DTE2 modifications, a pressure rise test of the GIS and linking of the additional core and ALARP areas to PSACS. This process was approved by the SSWP and required their formal notification on completion.

6.1. Lessons learned from DTE2

The lead time for production of the safety case and neutronics modelling placed some reliance upon the accuracy of operational planning for the DTE2 campaigns. Access requirements were subsequently impacted when experimental, operational and maintenance issues resulted in changes in the length and order of the operational campaigns and operational work patterns. Envisaging a short T–T campaign and a longer D–T campaign to follow without return to T–T operations, the restrictions in place for access during tritium operations were conservative and did not distinguish between T–T and D–T operations. This resulted in safety case modifications to enable access to the additional core/ ALARP areas, particularly to enable positive ion neutron injector maintenance, work on the neutral beam test bed, and to allow cryosystem repair work with substitution arrangements in place since T–T neutrons result in much lower direct radiation levels. With limited measurements or neutronics data available for the additional core and ALARP areas, safety case assessments were very conservative such that additional HP monitoring and controls were identified in point of work risk assessments. Temporary changes in restrictions also brought with them changes to work procedures and additional operator training to ensure awareness of the hazards.

When the tritium campaigns were completed, a further safety case modification was required to identify and define the key criteria under which a less onerous sub-set of designated safety systems and controls, commensurate with the diminished radiological hazards associated with non-tritium operations, could be applied. The criteria aimed to reduce mobile tritium, 14 MeV neutron levels, contamination levels and direct radiation levels to a similar range to those achieved post DTE1. However, in practice, clean-up operations were completed once tritium recovery from the vessel exhaust had reduced to levels at which further recovery by the AGHS was not considered practicable, and when the percentage of 14 MeV neutrons from D–T reactions had reduced to less than 5%. A target tritium concentration in the Torus exhaust of 0.02% had been set in the safety case modification for completion of post-DTE2 clean-up operations. From this point onwards, tritium would be recovered via EDS rather than via direct internal AGHS processing. In practice, once tritium levels started to stabilise below 0.08%, the decision of when to switch over the exhaust also had to take into account operational needs for the AGHS and Torus, including the requirements for the handling and storage of EDS condensate.

The only radiological fault sequences realised during the campaigns related to a very small leak of tritium into the GIS secondary containment and a separate ex-vessel coolant leak. The designated safety systems and plant indications operated as designed such that the GIS release was contained and purged into the AGHS buffer tank. The associated sub-system was isolated and subsequently repaired and returned to service. A small ex-Vessel LOCA occurred when an incorrectly installed coupling failed. The coolant was contained within Torus hall bunding. Both incidents were bounded by safety case assessments and designated safety systems operated successfully.

Data availability statement

The data cannot be made publicly available upon publication because no suitable repository exists for hosting data in this field of study. The data that support the findings of this study are available upon reasonable request from the author.

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