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ABSTRACT

Formal approval was required to permit tritium operation of the JET machine to proceed and a safety case was prepared in order to justify this. The standards and the methodology used are summarised both for the Active Gas Handling System and for the torus systems. The main design safety principles are described and accident sequences are reviewed. Both deterministic and probabilistic analysis was carried out and the main results in terms of source terms, on-site and off-site doses are presented. It is shown that operation of JET with deuterium-tritium plasmas complies with the ALARP principle.

1. INTRODUCTION

Deuterium-tritium (D-T) operation of the JET machine has always been considered as an integral part of the JET programme and was taken into account at the design stage particularly in the provision of massive biological shield walls and associated doors and shielding beams and in the completely remote controlled operational capability. Further preparations for D-T operation have been underway for a number of years, on a timescale consistent with the various extensions of the project. These preparations included the modification of a number of torus systems to ensure that they were compatible with tritium operation, the construction of the Active Gas Handling System (AGHS) [1] for recycling of tritium, the development of remote handling techniques for specific tasks and the preparation of a safety case for submission to the regulatory authorities.

A limited D-T experiment involving the supply of 0.1g of tritium to the machine and the production of $2x10^{18}$ neutrons was carried out in 1991 [2]. This Preliminary Tritium Experiment (PTE) required formal approval from the regulatory authorities which required the preparation of a safety case. As there was no significant off-site hazard and the on-site hazard was limited, a single Safety Analysis Report showed that with the provision of only limited safety systems, the safety targets were met [3]. This experiment provided valuable experience in the behaviour of tritium in machine systems and provided input for the updating of the safety case for the main D-T programme of JET.

The safety case leading up to D-T operation was submitted in several stages with the AGHS and torus systems being treated somewhat differently. In the case of the AGHS, which was designed explicitly for handling radioactive materials, the sub-systems of the whole plant could be subject to safety analysis before construction started. This could not be done for the torus as most of the systems were already in operation when that phase of the safety analysis started in 1994. In this case analysis considered the adequacy of the existing or planned systems and identified modifications required.

Tritium commissioning of the AGHS commenced in 1995 and was completed using 3g of tritium, by the end of 1996 [4] [5]. Approval was given in February 1997 for the DTE1 series of experiments which involved the fuelling of almost 100g of tritium into machine systems and the generation of 2.4×10^{20} 14MeV neutrons [6]. Although the safety case permitted up to 90g to be held on site and the generation of 5×10^{22} neutrons, lower management limits of 20g on site and 2.5×10^{20} 14MeV neutrons were set for DTE1. As tritium was recycled by the AGHS, these provided sufficient margin for tritium usage in the experimental programme and for D-T neutrons generated during clean-up operations.

DTE1 was completed in November 1997 and, following a period of operation in deuterium and hydrogen plasmas to facilitate tritium removal from the first wall [7], a shutdown took place in which remote handling equipment was used to exchange in-vessel components.

2. REGULATORY REQUIREMENTS AND STANDARDS

JET is situated in a UKAEA site and is bound by its statutes to comply with the safety standards of the UKAEA and to obtain the agreement of the UKAEA Director of Safety before each stage of radioactive operation. The standards are equivalent to those which would apply to a site licensed under the Nuclear Installations Act 1965 although, as a fusion facility, JET does not come under this legislation. Amongst other health and safety legislation, JET is subject to the Ionising Radiations Regulations 1985 and the Radioactive Substances Act 1993 (RSA93) through which official approval for radioactive holdings and waste disposal is granted by the Environment Agency (EA). The standards for a nuclear licensed site are laid down in the Safety Assessment Principles (SAPs) published by the Health & Safety Executive (HSE) [8] in 1992. As well as guidance on engineering principles, they set down criteria for evaluation of accident scenarios and routine operations in accordance with mandatory Basic Safety Limits (BSL) and Basic Safety Objectives (BSO) which, if achieved, can be used to justify that the ALARP principle is met.

The SAPs include both deterministic and probabilistic criteria, the latter being shown in Table 1. These were supplemented by internal UKAEA standards where appropriate.

Table 1: SAP Dose Bands

Off-site Dose Range(mSv)	**Total Predicted Frequency**	
	BSL	**BSO**
>1000	10^{-4}	10^{-6}
100-100	10^{-3}	10^{-5}
10-100	10^{-2}	10^{-4}
1-10	10^{-1}	10^{-3}
0.1-1	1	10^{-2}
0.01-0.1	*10*	*10^{-1}*

Most of the AGHS analysis was carried out before the 1992 SAPs were published and used UKAEA probabilistic criteria [9]. However, as shown later the AGHS also complies with the later standards.

In addition to the externally imposed standards, JET established criteria for DT operation as follows:

i) DT operations should have no significant effect on the local environment;
ii) The annual dose to classified workers should be < 5 mSv and to others <1mSv;
iii) There should be no requirement for emergency action to be taken off site.

The first criterion was interpreted as an annual dose of < 50μSv off-site and was considered in the submissions to the EA when seeking RSA93 authorisations. The assessment of the dose to the most exposed individual showed that it would not exceed 17 μSv for the discharges listed in Table 2. For the production of 5×10^{22} neutrons per year, the additional dose at the site boundary due to direct γ radiation, skyshine and neutron leakage was less than 5μSv. The second criterion guided the design of the tokamak shielding arrangements particularly for the case of the many torus hall penetrations which had been provided to accommodate diagnostics and other enhancements. The third criterion set a bound on the consequences of accident scenarios and thus for example limited the mobilisable inventory of tritium. It also meant that much of the range of the SAP probabilistic criteria did not apply and therefore, additional assessment criteria were included for the JET safety case. These can be used to demonstrate that there is not an excessive risk from low dose accidents and are shown as an additional band in italics in Table 1.

Table 2: Discharge Authorisations

Discharge Route	Activity	Annual Discharge Limit	Critical Group Dose
Aerial	Tritium as oxide	90TBq (2500 Ci)	6.3μSv
	Tritium (excl oxide)	110TBq (3000 Ci)	<1μSv
	Activated air and coolant	24TBq (650 Ci) total βγ	7μSv
	Activated dust	1GBq	<1μSv
Liquid(Aqueous to Thames)	Tritium	10TBq	0.12μSv
	Activation products	100MBq	<1μSv

In addition, the SAP definition of the upper dose limit of 100mSv for Design Basis Accidents (DBA) was not appropriate as JET was considered as a “Category 2” plant under the UKAEA definitions. This was equivalent in hazard potential to a small research reactor. To match the categorisation of JET; to provide a more rigorous demonstration that risks to the public are as low as reasonably practicable; and to justify fewer levels of redundancy and diversity for engineered safety systems than would be the case for a power producing nuclear reactor, an off-site dose limit of 50μSv was set for DBAs with an upper limit for severe accidents of 5mSv.

3. FORMAT OF SAFETY CASE

The prime documents in the separate safety submissions for the AGHS and torus were:

i) Preliminary Safety Report (PSR);

ii) Pre-Construction (or Commencement) Safety Report (PCSR);

iii) Pre-Commissioning Safety Report (PCmSR).

These documents were supported by a number of reports on topics such as hazard identification, dispersion modelling etc.

The purpose of the two PSRs was to establish the safety principles and guidelines, identify the significant hazards which would arise during D-T operation, set down the principles for engineered safety systems and barriers identified at that stage, and summarise the proposed arrangements for radiation protection, waste management and management of safety. Acceptance of each PSR then led to more detailed hazard identification and analysis

A PCSR would normally be prepared before constructing or modifying a plant. In the case of the AGHS, to enable construction to start before the detailed design was complete, this assessment took the form of a number of separate "Design Safety Reviews" (DSRs) and a PCmSR was produced prior to tritium commissioning. For the torus, because JET was already in operation with deuterium (D-D) plasmas and the analysis was mainly retrospective, the PCSR and PCmSR were effectively the same document.

3.1 AGHS Safety Case

The AGHS DSRs comprised a Failure Mode and Effects Analysis (FMEA) which identified the consequences of malfunctions and the actions of protection systems to show deterministically that the consequences of plant faults were acceptable, and a probabilistic analysis in which failure rates and availabilities were considered. The analysis for the AGHS was able to draw on the experience of other tritium handling facilities, supplemented where necessary by specific tests.

The basic failure which applied to all accident scenarios which were analysed in detail was loss of primary containment either adventitiously or as a result of another failure such as overheating of a uranium bed. The initiating events were analysed for each sub-system by Fault Tree analysis and event trees were constructed to enable the effect of mitigating systems such as secondary containment and the Exhaust Detritiation System (EDS) to be evaluated. A separate DSR was prepared for each major subsystem of the AGHS. The basis for acceptance by the regulator was that the FMEA should show that the consequences of failure were acceptable and the product of the frequency and release from any accident sequence should be less than 0.37TBq/

year. This figure, which had also been used in the PTE, was chosen in consultation with the regulator as a reasonable compromise to enable design to proceed on a sub-system by sub-system basis with the expectation that the overall risk from the plant would be acceptable [9]. Approval for procurement was only given when the design and the process for determination of unresolved safety issues had been accepted by the regulator.

Because the DSRs were carried out individually, care was needed to ensure that interface and common issues were properly addressed. They were therefore supplemented by safety reviews of the control system, safety interlocks, sub-system interfaces and a building review covering the common plant. The latter included consideration of seismic issues which because of the low seismicity of the UK was based on a "walk-down" approach in which restraints, if necessary, were retrofitted to the plant. The AGHS PCmSR brought together all of the DSR accident sequences and derived the overall risk, including that from external events. Management and commissioning arrangements were also covered. The AGHS safety case is further considered in refs [10] and [11].

3.2 Torus Safety Case

One of the difficulties faced in the preparation of the torus safety case was that (in contrast to the AGHS) there was no comparable plant in operation anywhere, the TFTR licensing process [12] being at approximately the same stage at the time of the PTE. Normally for a nuclear plant, the hazards are well known and a set of well-substantiated reference accidents can be presented for analysis. As JET had no previous D-T operation to draw on apart from the PTE [3], justification of the selection of accidents for analysis, and in particular the completeness of the identification process was demanded by the regulators.

However, the hazard identification was able to draw on the experience of the operation of JET and other fusion machines in deuterium to identify potential accident sequences, particularly relating to vacuum systems. An extensive database of vacuum leaks and related events had been built up at JET, which enabled the type and frequency of these events to be categorised.

To carry out a detailed analysis of all JET systems would have been an overwhelming and largely unproductive task given that most JET systems such as pulsed power supplies do not create any direct radiological hazard. In order to minimise analysis of these events which were of no safety significance, a screening process was therefore carried out before the PCSR commenced. This examined the potential radioactive inventory of each system and the personnel and public dose which could arise from release of the maximum mobile inventory, assuming credit only for the torus hall and its containment. Systems in which the potential dose from any accident including severe hypothetical accidents did not exceed 20μSv off-site were deemed to be "non-hazardous" and were not analysed further except where they had the potential to create external events in other systems. This enabled much of the "conventional" JET plant such as power supplies to be eliminated from detailed analysis.

A complementary process to the screening was a HAZOP (Hazard and Operability) study which is a structured method of analysing the consequences of system deviations from normal and then identifying accident initiators [13]. It is extensively used in the chemical industry and was considered by the regulators to be an essential process to demonstrate the completeness of the hazard identification. It was carried out in a number of formalised sessions involving the relevant JET designers, operators and safety analysts.

As well as producing a list of events for analysis, the HAZOP study identified some areas where engineering modifications would be required before tritium operation and prompted a number of accident sequences to be examined at an early stage to determine if they would be significant.

Approximately 150 initiating events were examined in the torus analysis and were categorised into groups as follows:

1) Loss of vacuum accident (LOVA);
2) In-vessel loss of coolant (fluid) accident (in-V LOCA);
3) Ex-vessel loss of coolant (fluid) accident (ex-V LOCA);
4) Loss of flow accident (LOFA);
5) Plasma fuelling and heating system events (PFHSE);
6) Magnet events (ME);
7) Shielding events (SE);
8) Loss of plasma control (LOPC);
9) Loss of electrical power (LOEP);
10) External events (EE).

To simplify the analysis, many events were considered to be bounded by the consequences of key events such as loss-of-vacuum accidents (LOVA) so the number of fault sequences analysed in detail were somewhat less than in the AGHS case.

The torus PCmSR consisted of essentially four parts:

a) A detailed description of the JET facility as analysed for the safety case. The purpose of this was several fold:
   i) To provide sufficient information for peer reviewers to understand the JET plant;
   ii) To provide evidence to justify the “non-hazardous” nature of certain plant systems;
   iii) To provide a reference design from which deviations would be subject to a formal modification procedure;
   iv) To define aspects of “management of safety” related to waste management, health physics and operational procedures.

b) The justification for calculations of source terms and consequences in the accident analysis, for example dispersion and dose uptake modelling;

c) A deterministic analysis of the consequences of identified events and assessment against pre-determined standards;

d) A probabilistic analysis of accident consequences and frequencies, and comparison against pre-determined standards.

The analysis carried out under c) and, d) was an iterative process in which relatively conservative analysis was initially carried out. Where certain accident scenarios did not meet the required standard, more refined engineering analysis was carried out to reduce the margins of conservatism needed. In some cases, plant modifications were ultimately required in order to satisfy the risk and/or dose targets applied in the safety analysis.

4. DESIGN SAFETY PRINCIPLES

Over 30 Design Safety Principles were identified in the PSRs, which determined the design features of the AGHS and led to modifications to the torus systems during the preparation for D-T operation.

The most important principle from an engineering point of view was that no single failure of any system should result in doses to workers or members of the public greater than the annual legal limits. This had many effects including the modification of safety systems for drain down of water and Freon [14], the enclosing of all systems containing significant tritium in secondary contaminants and the conversion of all vacuum windows and most feedthroughs to a double arrangement with testable interspace [15].

An essential principle for routine operation was that the dose to non-classified workers should not exceed l mSv/year. This required massive biological shielding, designed in principle for $5x10^{23}$ neutrons/year, a comprehensive access control and interlock system and a system of ventilation which provided graded depression in the active areas and discharge of activity from controlled stacks [16]. Modifications to shielding and in particular penetration sealing, has been assessed against the likely maximum neutron production of up to $5x10^{22}$/year.

All systems with the potential to exhaust significant quantities of tritium were required to be vented through to the AGHS via vacuum crowns or atmospheric pressure exhaust lines. In both cases, the Exhaust Detritiation System (EDS) in the AGHS would be automatically brought into service in the event of a vacuum failure or significant tritium release inside the vacuum vessel. The EDS is one of the key safety systems on JET as the final barrier preventing tritium release to the environment [17]. HTO and HT are collected as tritiated water on molecular sieve beds. The beds can be regenerated to recover the tritium in water which can be drummed for orderly disposal or recovery (Figure 1).

A further principle was that key safety-related protection and interlock systems should be hardwired where possible. This avoided the difficulties associated with determining the failure rates and reliability of software based systems.

Further design safety issues were identified during the course of conducting the safety analysis and led to changes to the design of the AGHS and modifications to torus systems in an iterative way. In some cases, the analysis showed that no changes were needed. In other cases,

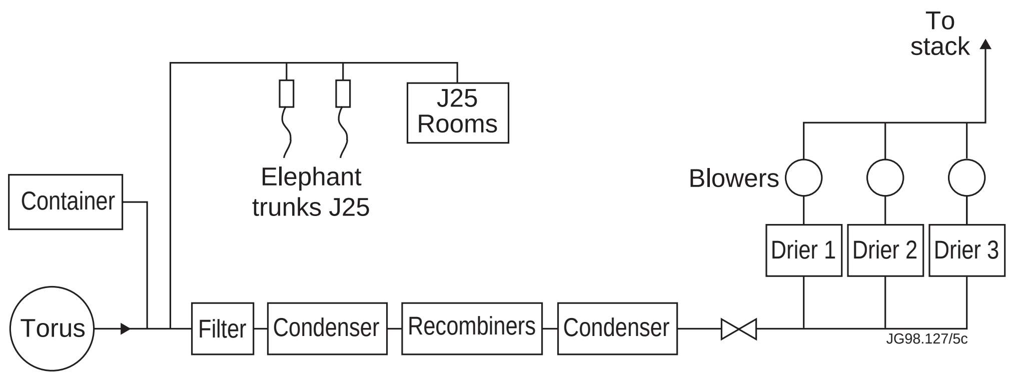

Fig.1: Exhaust Detritiation System

the need was identified to perform experiments to improve the understanding of the likely course of accidents. These included:

i) The possibility of runaway carbon tile temperature excursions in the event of air ingress was considered and shown to be improbable [18].

ii) The possibility of a fire in the torus hall was considered. Even with relatively low integrated neutron production, the dose-rates in the torus hall immediately after pulsing would be high (5mSv/h) and would preclude access for fire-fighting. A CO_2 fire suppression system could not be used as this was incompatible with the torus hall overpressure capability and both this and a Halon based system conflicted with the requirement to maintain the torus hall under depression. The solution adopted was to provide partial inerting of the torus hall atmosphere by reducing the oxygen content to 13-15% to suppress normal combustion [14] [19].

iii) The need for fail-safe operation of turbopump bypass valves to ensure that in the event of a power failure the torus overprotection system would remain functional. This aspect also required the provision of a fail safe “open-path” through the EDS.

iv) U-Bed air ingress. The reaction of air with the pyrophoric uranium powder was well known and tests and analysis were required to demonstrate the safety of uranium for storage and transfer of tritium within the AGHS [20] [21]. The use of nitrogen gloveboxes and the existence of argon and hydrogen blanketing effects plus the provision of overtemperature protection and triple containment permitted an acceptable safety justification to be made [1].

Activation source terms were based on the prediction of 5×10^{22} D-T neutrons in one year and a maximum of 10^{20} neutrons in one pulse. Activation calculations were primarily based on the work of Avery who used the ORIGEN code and UKCTRIIIA library [22]. In view of the limited number of neutron groups in original analysis, cross-checks were carried out using the FISPACT or REAC codes with EAF3 or ECN-4 activation cross-section libraries. Differences were generally insignificant but where appropriate, the most conservative results were used. The main activities and quantities of mobilisable activated material are listed in Table 3.

Table 3: Activated Material

	Quantity	Main Activity	
Freon 113	95×10^3kg	S^{35}	4.4TBq
		P^{32}	4.2TBq
SF6	$30m^3$	P^{32}	0.03TBq
Halon	5×10^3kg	$Br^{80}m$	2.1TBq
Tokamak dust	1kg	Co^{60}	0.02TBq

From measurements made at shutdowns, around 1kg of dust was considered to be present in the vacuum vessel [23]. The composition could include Inconel, carbon and beryllium. A variety of nuclides including Mn^{54} and Co^{58} could be present in this dust, depending on its composition. Because of the low air change rate of the torus hall atmosphere (once every 12 hours) short lived activity present after a pulse could be discounted in off-site dose assessments. To allow sufficient conservatism and to simplify the analysis, the bounding assumption was made in many accident sequences that all the activity was Co^{60}.

The activation of other fluid such as cryogens, baking plant gas and air was insignificant from the accident analysis point of view but needed to be considered in terms of the radioactive waste authorisations. Other solid activated material such as coil epoxy was eliminated from the accident analysis as the dose resulting from the release of the entire inventory was estimated to be less than 20μSv.

The tritium source terms are effectively independent of the neutron production and are a function of the tritium throughput. The main source terms were:

i) AGHS and Supply Systems

The AGHS was designed to handle up to 30g of tritium per day and is authorised to store 90g. The tritium inventory is distributed in plant subsystems with the highest proportion being located in the Cryogenic Distillation (CD) system [24] which requires 35g inventory for full operation. The supply lines from the AGHS have a significant volume in terms of tritium (3.2g) and are normally pumped back to AGHS at the end of each day. These values determined the source terms for AGHS accidents. As a result of double containment, release fractions for most accident sequences were of the order of a few TBq.

ii) Cryogenic surfaces

A total of 10g was assumed to be trapped on LHe and LN_2 panels on torus and Neutral Beam (NB) cryopumps.

This is governed by the tritium fuelling requirements and the frequency of regeneration and is basically a management limit. In tritium operation, the helium panels of the cryopump of the Neutral Injector Box (NIB) which supplied the tritium beams [25], and of the torus cryopump, were regenerated by warming to 77K. This released the tritium inventory which was then pumped back to the AGHS. In this manner the inventory on the pumps was kept below the management limit. All the tritium on cryopumps is assumed to be released into the vacuum envelope during a serious LOCA or LOVA.

iii) First wall and other solid material

Tritium may be both co-deposited with carbon or other first wall materials or diffused into the bulk material. The total amount is governed primarily by the requirements for wall loading to obtain the correct fuelling ratio. Up to 20g was assumed to be retained with ~2g released into the vacuum vessel in an air ingress accident [26]. In practice, during DTE1, there was never more than 11g of tritium retained in the torus [7].

iv) Permeated tritium

Chronic tritium releases as a result of permeation were considered in the application for waste disposal and were estimated to be up to 100TBq/year [27]. The amount held up as releasable tritium in the torus vessel baking loop was assessed as 1TBq.

In the above cases, the ratio of HTO to HT is determined by the accident sequences. In fires, 100% conversion to HTO is assumed whereas for releases into the vacuum vessel and from the AGHS, a 10% conversion over 12 hours is assumed.

6. ENVIRONMENTAL PATHWAYS AND DISPERSION

The most significant route for off-site exposure from accidental releases of activity is by atmospheric discharges. The prediction of atmospheric dispersion and the resultant environmental concentration of radionuclides was carried out using Gaussian plume models. Three locations, torus hall roof, torus building ground level and AGHS building stack, were considered for 1 hour and 12 hour release durations under the six Pasquill stability criteria including C and D with rain [28]. Inhalation, skin absorption and radiation doses for tritium and about 40 activation products were calculated at distances from 100m to 1km and the ingestion dose to a hypothetically exposed individual adult determined, assuming food products were obtained from 1km downwind of the release. The food groups included milk, grain, leafy vegetables, root vegetables, beef and lamb.

The inventory of mobilisable activation products was low and the maximum dose from an accident sequence involving activation products (Freon release) was 260μSv at the site boundary the nearest point of which was at 250m (figure 2).

The inventory of tritium was however significant and more rigorous modelling was necessary to permit realistic operating limits to be established. Tritium doses were modelled

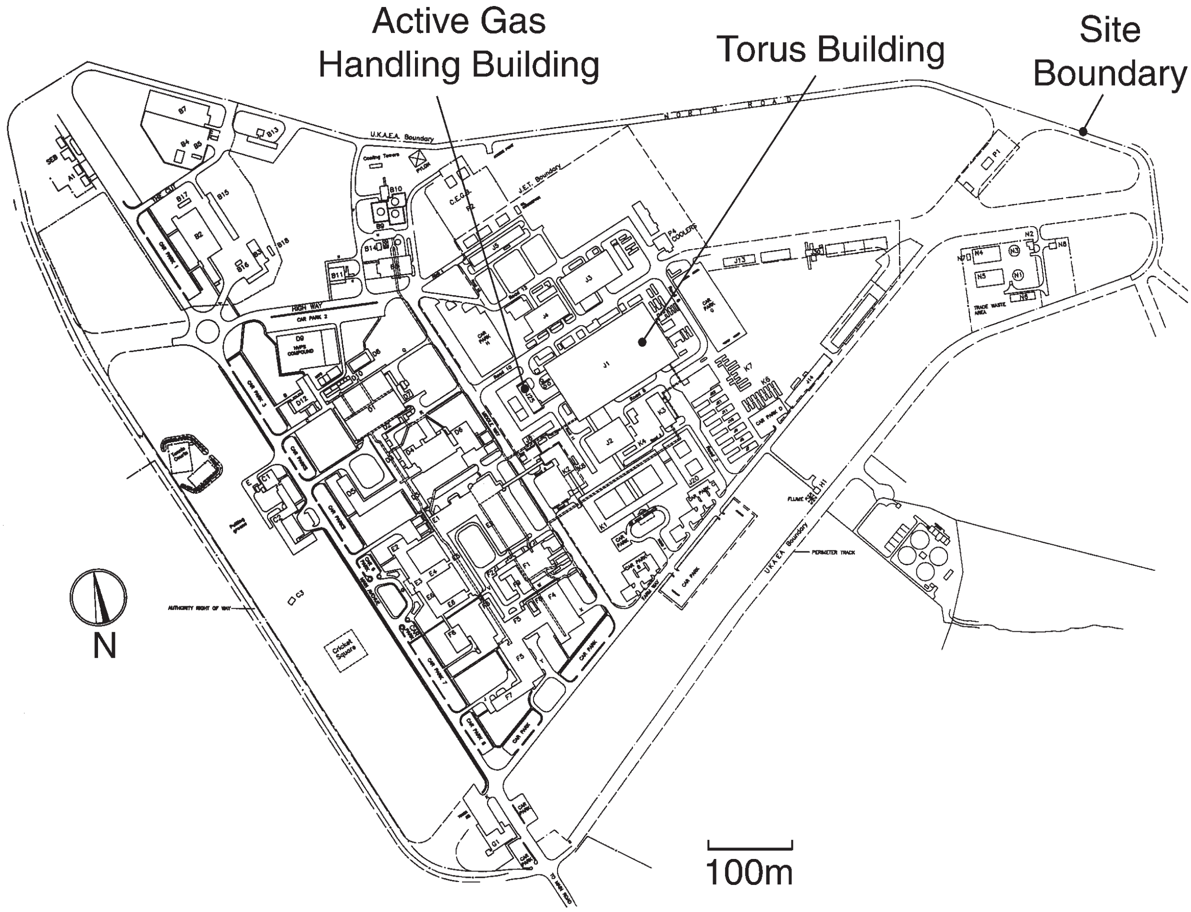

Fig.2: JET Site Boundary

separately using the ETMOD [29] code which has been validated against the results of field tests to take into account tritium in the form of both HT and HTO, the effects of soil and vegetation absorption and re-emission, and conversion to Organically Bound Tritium (OBT).

The time period over which contaminated food was ingested was dependent on the characteristics of the food group and varied from one week for leafy vegetables to six months for meat assuming no loss on storage. Consumption of foodstuffs up to the notional time of harvest (1 day after a release) was also taken into account.

The worst-case doses used in the accident analysis were for ground level releases over one hour and were 4.2μSv/TBq at 250m for HTO releases under Pasquill class F conditions. In comparison elevated releases were highest under class A conditions where building wake entrainment contributed to an HTO dose of 1μSv/TBq. Doses arising from HT releases were dominated by the effects of reemission and conversion to HTO and were 20-40 times lower in the case of inhalation and skin absorption. The maximum dose for ingestion at 1Km was 0.8μSv/TBq for HTO releases and 0.1μSv/TBq for HT releases.

Although some tritiated hydrocarbons are created by reactions with the vacuum vessel first-wall, these were not modelled separately and were assumed to have the same characteristics as HTO.

7. ANALYSIS IN AGHS PCMSR

The detailed analysis was carried out in the individual DSRs which identified around 400 accident sequences, quantified the amount of tritium released and made an estimate of the annual frequency. With the highest dose off-site being assessed as 4mSv, the results were shown to satisfy the deterministic safety principles.

Operator doses were determined using two cases: a rapid, essentially instantaneous release in which the operator is exposed to a very high tritium concentration for 20 seconds; and a more realistic slow release over a period of 1 hour. The first case was hypothetical except in the case of an explosion or other external event which destroys both the primary and secondary containments. Assuming instantaneous conversion to HTO gave hypothetical doses in the Sv range but allowed the upper bound of risk to workers to be conservatively quantified at 2×10^{-5} per year. The slow release scenario for about 100 accident scenarios which resulted in a tritium release into a potentially occupied area gave 4×10^{-7} per year. These values may be compared with the BSL of 10^{-4} and BSO of 10^{-6}.

In accordance with the UKAEA standards which were then current, the overall risk on and off-site was calculated. Using a value for premature death of 4×10^{-2}/Sv gave a value of 5×10^{-7} and 1.1×10^{-8} respectively per year, taking account occupancy factors and variability in wind direction compared with the UKAEA standard of 10^{-6} per year.

AGHS accident sequences may also be compared with the dose bands in the SAPs and this comparison is laid out in Table 4.

These results, which do not include a factor to allow for variability of wind direction, show that the AGHS complies with ALARP without further justification.

Table 4: Comparison of AGHS results with SAPs

Off-site Dose Range(mSv)	**Total Predicted Frequency**		
	BSL	**BSO**	**JET AGHS**
1-10	10^{-1}	10^{-3}	4×10^{-5}
0.1-1	1	10^{-2}	1.4×10^{-3}
0.01-0.1	10	10^{-1}	4.1×10^{-2}

8. ANALYSIS IN TORUS PCMSR

8.1 Deterministic Analysis

Accidents were classified into “Design Basis Accidents” (DBA) which had been considered in the design of the plant and for which specific provisions had been made to minimise the consequences and “Beyond Design Basis Accidents” (BDBA) which were considered to be sufficiently improbable that no specific mitigating systems were considered to be necessary. In the latter case, however, where systems existed, credit could be claimed for them in the analysis. The analysis was required to show that for DBAs, the maximum dose off-site was <50μSv and for BDBAs <5mSv.

On-site doses were calculated at 100m and within the accident "room", taking into account, where appropriate, the concentration in the local breathing zone and a realistic time to evacuate (15 mins). It was required to show that for DBAs, on-site doses were <5mSv, which was the JET management limit for annual radiation exposure.

Apart from those arising from catastrophic external events and hypothetical plant events, single failures in which only the primary system failed, or in which no credit was taken for protection systems, were classed as DBAs. Events in which the design action functioned were by definition DBAs and this included the many protection systems which were already in place during DD operation. Any protection system which was required to function in order to comply with the DBA dose limit was defined as a safety mechanism. Common cause failures were treated in the same way. Dual failures in which either one or more of the safety mechanisms failed in addition, or was unavailable, or in which independent systems failed were generally classed as BDBAs (See figure 3).

The main accidents considered in the PCSR are summarised in the following sections.

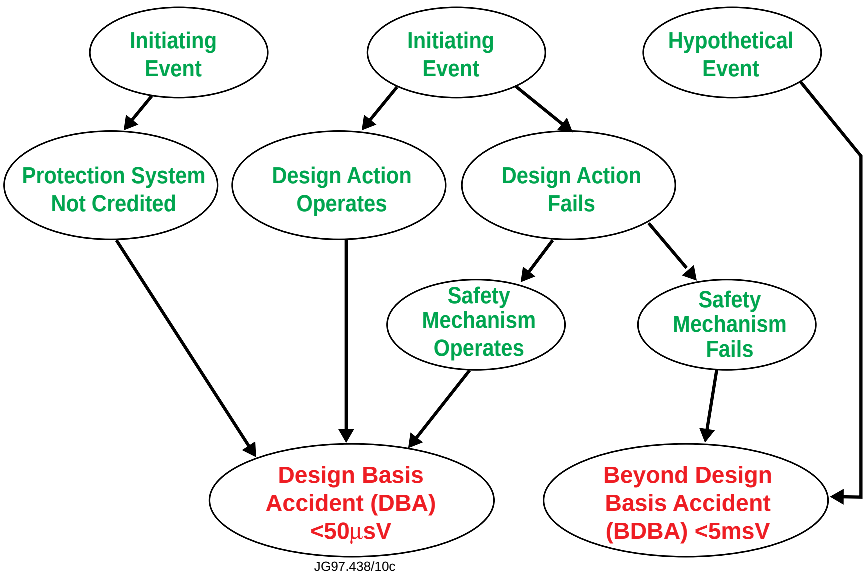

Fig.3: Classification of Events

8.1.1 LOVA

Normally, a vacuum leak would be detected at the stage where it affected plasma conditions and the operator would follow procedures to make the plant safe. If the leak developed quickly, automatic action would take over. At 200mbar vessel pressure, the torus turbomolecular pumps would automatically be bypassed. The AGHS cryopumps would then be isolated and pumping taken over by the mechanical forevacuum system. At the same time, the Exhaust Detritiation system would be connected into the torus pumping line via a non-return valve in readiness for providing depression in the torus at a maximum throughput of 500m^3/h if the pressure reached the ED inlet pressure of around 900mbar. Air entering the torus will be heated through contact with the vessel walls at 320^{o}C and in the absence of EDS ventilation, a fraction would be expelled back through the breach. For the DBA size of 50mm equivalent diameter, it has been shown that no efflux (other than by diffusion) occurs. Tritium released from torus and NIB cryopumps and from the torus walls is trapped on the EDS molecular sieve after oxidation. This DBA single failure case led to a predicted dose of 23μSv.

More severe accidents considered included a double-ended guillotine fracture of the main vacuum line between the torus and AGHS; with the alternative line to the AGHS from the NIBs being unavailable, leading to an off-site dose of 1.2mSv (ground level release) It is inconceivable that such a failure could occur in a 300mm diameter thick-walled pipe with welded or substantial flanged joints. A dual failure case where the EDS oxidation catalyst operates when the EDS is required to ventilate the torus following a LOVA but the molecular sieve fails to collect the tritiated water resulting in a dose of 4.9mSv was considered as a bounding case.

8.1.2 In-vessel LOCA

There are a large number of components within the JET vacuum boundary, which are supplied with fluids for cryogenic or heat removal purposes. Both the cryogenic and demineralised water supply systems contain sufficient fluid (>500kg) to pressurise the vacuum vessel unacceptably. To provide a controlled discharge path and a means by which the EDS could provide depression after any surge of pressure, a bursting disc rated at 50mb is backed by a flap valve set at 200mb differential. Rupture of the disc was however considered as a BDBA and to prevent this, protection systems were necessary on the in-vessel cryogenic and water systems [14]. These needed to take into account the action of the automatic drain-down system in the event of vacuum leaks, the need to avoid freeze-up and possible failure of components and the ability to carry out the normal start-up and shutdown operations in an orderly fashion. Major modifications were required to the control systems to limit the amount of water capable of being discharged so that the pressure could not rise above atmospheric.

In addition to the LOVA actions, the water systems are isolated at l5mbar vessel pressure, each individual loop tested for leakage, and the leaking loop drained to a tank/header system held at 100mbar so that water drains by gravity. The control of the draining sequence is via a hardware system. The functions of this hardware system are duplicated by a Programmable Logic Controller (PLC), which also executes other functions for more intelligent handling of lower level LOCA and LOFA events [14]. Protection systems also exist to limit the amount of water capable of entering a NIB and for LN_2 and LHe supplies, these are described in more detail in reference [14].

The analysis showed that following a large water pipe rupture up to 33kg of liquid may enter the vessel prior to isolation valve closure with 71kg of liquid and 11kg of steam subsequently discharged. The transient pressure would not cause the rupture disc to fail and the long-term steam production ratio would be easily handled by the EDS. The effect of steam/graphite reaction was shown to have a negligible effect on the outcome. Doses were similar to those calculated for LOVA.

Failure of the protection system resulting in continued pressurisation of the vessel would result in rupture of the bursting disc and discharge in a controlled manner via the basement stack. This would lead to a rapid venting of all the mobilisable tritium in the machine with slow conversion to HTO in the presence of steam resulting in an off-site dose of 180μSv. In the event that an operator was present in the basement at the time of discharge, the dose received would be 37mSv.

The in-vessel Freon circuits, for the divertor coil cooling, were all double-contained with a pumped interspace with pressure detection. Thus an in-vessel LOCA involving Freon release was considered as a BDBA.

8.1.3 Ex-vessel LOCA

The main concern in this case is the Freon used for cooling the toroidal field and divertor coils. Activated vapour would be released by flashing, boiling or evaporation depending on the location of a leak. The external storage tank is passively protected by a bund containing a pool of water to inhibit evaporation. Spills in the torus hall are guarded against by a Freon detection system backed up by expansion tank level measurement either of which can trigger an automatic drain-down of the system [14]. Apart from the cases where the Freon spills into pumped gullies or bunds, the analysis assumes that it all evaporates in a period of 12 hours leading to a DBA maximum off-site dose of around 30μSv and BDBA (in the case of drain-down failure) of around 250μSv.

8.1.4 LOFA, ME, LOPC and LOEP

With the low stored energy of JET systems, loss of flow accidents would not result in consequences beyond those potentially arising from LOVA and LOCA. In the other cases, the main consequence is a potential plasma disruption, the consequences of which were bounded by LOVA.

8.1.5 PFHSE

The main events considered under this category were those associated with tritium and hydrogen isotopes released from fuelling systems. The possibility of detonation and damage to other systems was also considered. Other events were generally bounded by LOVA and LOCA. For example, failure of NB control and protection systems could conceivably lead to penetration of the vessel wall or cooling pipe failure. In general, these would involve multiple simultaneous failures and were considered as BDBAs [14].

Because of secondary containment systems which were ventilated positively to AGHS buffer tanks with protection provided by the EDS, DBAs in the tritium supply and distribution systems were concluded not to lead to off-site doses. It was conceivable that an external event could cause simultaneous failure of primary and secondary containment. Such dual failures associated with deflagration would result in conversion of all the tritium in the supply lines to oxide from an off-site dose of 1.lmSv.

8.1.6 Shielding Events

Interlocking prevents pulsing taking place when certain removable shielding elements are not in place. Others are controlled by administrative procedures. In addition, there are strategically placed gamma monitors and the assumption made is that only one pulse could take place without the correct shielding. It is not considered credible that full performance pulsing could take place with both torus hall shield doors open. Even in this case the off-site dose would be a few mSv. It is considered credible that both shield doors could be open with the machine fully activated at the end of a DT campaign, but the off-site dose (~1μSv) in this case is insignificant.

Shielding events do however have potential for operator doses >1Sv and the standard of the controls reflect this rather than the public exposure aspect.

8.1.7 External Events

Events considered included seismic, aircraft crash and dropped loads. Most were bounded by previously analysed events but the highest dose would arise from a large aircraft crash which mobilised the total site inventory of tritium and Freon (7mSv). Given the low probability of this event, and the likely non-radioactive consequences, this was considered acceptable even though it slightly exceeded the off-site dose limit.

8.2 Probabilistic Analysis

The purpose of probabilistic analysis was to show that the overall risk to workers and members of the public from D-T operation is acceptable and meets the criteria set down in the NII Safety Assessment Principles. It involved consideration of not only DBAs but also of BDBAs and lesser accidents, which were considered in the deterministic analysis as, bounded by the DBAs. Failure rate data drew on both JET operational experience from 1983 onwards, data related to fusion applications [30], and published data normally used for fission plant analysis. Much of this data is derived under different operating conditions compared to JET. For example failure rates of high pressure nuclear systems may be grossly pessimistic when applied to vacuum systems at JET but this provides weight to the conservatism of the analysis. The actual JET reliability data collected over the 12 years of D-D can be validly extrapolated to D-T operation as JET is not designed to be a net energy producer even with the most optimistic performance in the D-T phases and stresses arise mainly from establishing the conditions for and from failure of plasma confinement and are all controlled to be within the mechanical design limits of the plant. The JET failure data was particularly relevant in the case of the LOVA sequences which dominated the overall risk. Data was available for windows of various sizes, feedthroughs and other vacuum components although in some cases where no failures had occurred during the lifetime of JET, a failure rate of 1/3 in the time at risk was assumed. The catastrophic failure rate was assumed to be 1/10 the leakage rate and for double components (eg windows) a β factor of 0.2 for common-cause failure was used.

Taking 20μSv off-site as the 'non-hazardous' dose, the mission time of the EDS to detritiate to this level was assessed to be less than the time taken to fully load one of the three molecular sieve dryers. Dryer change over and regeneration failure was therefore not considered in the EDS availability calculations.

All the accident sequences discussed in section 8 above were subject to probabilistic analysis. The frequency of each accident sequence was calculated from the probability of the initiating event and the availability of mitigating systems such as EDS. The sequences were then grouped into the off-site dose bands set down in the SAPs plus the additional "JET" band and summated to enable them to be compared with the BSO and BSL. The results are given in Table 5.

The frequency of all events giving doses to a member of the public in the range of 10 to10^2μSv is dominated by LOVA (55%) and LOPC (32%). Ex-vessel LOCA contributes about 8% to this range. The remaining events contribute the remaining 5% and can thus be considered insignificant. The total frequency in this dose range lies about a factor of 20 below the BSL, and therefore is considered acceptable. There is considerable pessimism in this analysis; particularly for LOPC where it is assumed that any major disturbance to the magnetic field plasma heating or plasma density results in a disruption. Historical JET data is then used to assign a probability to a resultant LOVA or in-vessel LOCA.

The total frequencies for the two higher dose bands (10^2 to 10^3μSv and 10^3 to 10^4μSv) lie below the HSE BSO levels and can therefore be justified as ALARP.

At JET, doses to a member of the public are generally below 5.0 x 10^2μSv, which accounts for the heavier weighting of frequencies in the two lower dose bands. Doses are low because the source terms are small (compared to nuclear plants) and releases are well dispersed. There was also a deliberate effort made to over predict single failure frequencies and, because the events cause doses in the 10-20μSv range, the lower dose band frequency became disproportionately larger.

The frequencies in the torus analysis are about an order of magnitude higher than that of the AGHS. It could be concluded that this reflects the fact that the AGHS was designed hand-in-hand with safety analysis. However it more likely is attributable to the simplifying conservative

Table 5: Total Frequency of Accident Categories Grouped in Radiological Dose Bands

Event	Dose Band		Frequency (1/y)	Band	
	Minimum (μSv)	Maximum (μSv)		BSO (1/y)	BSL (1/y)
LOVA	*1.00E + 01	1.00E + 02	2.87E-01	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	9.00E-03	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	4.41E-05	1.00E-03	1.00E - 01
Ex-Vessel	1.00E + 01	1.00E + 02	4.23E-02	1.00E-01	1.00E + 01
LOCA	1.00E + 02	1.00E + 03	2.70E-05	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	2.20E-08	1.00E-03	1.00E - 01
LOFA	1.00E + 01	1.00E + 02	6.52E-03	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	2.14E-06	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	1.02E-06	1.00E-03	1.00E - 01
PHFSE	1.00E + 01	1.00E + 02	9.75E-04	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	4.23E-05	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	1.40E-07	1.00E-03	1.00E - 01
ME	1.00E + 01	1.00E + 02	1.70E-02	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	7.48E-05	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	0.00E + 00	1.00E-03	1.00E - 01
SE	1.00E + 01	1.00E + 02	1.10E-04	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	1.10E-04	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	1.22E-04	1.00E-03	1.00E - 01
LOPC	1.00E + 01	1.00E + 02	1.68E-01	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	1.24E-04	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	2.60E-05	1.00E-03	1.00E - 01
LOEP*	1.00E + 01	1.00E + 02	0.00E + 00	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	0.00E + 00	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	0.00E + 00	1.00E-03	1.00E - 01
EE	1.00E + 01	1.00E + 02	1.00E – 03	1.00E-01	1.00E + 01
	1.00E + 02	1.00E + 03	5.25E – 06	1.00E-02	1.00E + 00
	1.00E + 03	1.00E + 04	1.00E – 04	1.00E-03	1.00E - 01
Total =	1.00E + 01	1.00E + 02	5.23E – 01	1.00E + 01	1.00E + 01
Total =	1.00E + 02	1.00E + 03	9.38E – 03	1.00E + 02	1.00E + 00
Total =	1.00E + 03	1.00E + 04	2.94E – 04	1.00E + 03	1.00E - 01

*Note: In-vessel LOCA events are incorporated into LOVA and LOPC. LOEP events are incorporated into LOFA and LOPC.

assumptions used in the torus analysis. The AGHS, being constructed from a number of sub-systems, which from a safety viewpoint were conceptually similar, was more amenable to rigorous analysis than the torus with a variety of disparate systems which were best analysed using bounding cases

On-site doses were similarly calculated and, using a risk of premature death of 4 x 10^{-2}/Sv, compared against the BSO and BSL as shown in Table 6. It can be seen that virtually all are below the BSO. The highest risk arises in the torus building basement but as no account is taken of occupancy factors, this is considered to be ALARP.

In this case, the torus and AGHS risks are broadly similar.

Table 6: Summary of Radiological Risks to JET Personnel from Accidents at JET

Event	Location	Risk (1/y)	NII Targets	
			BSO (max)	BSL (max)
LOVA *	100m	1.32E-07		
	Inside	1.09E-05		
Ex-vessel LOCA	100m	4.27E-07		
	Inside	1.92E-06		
LOFA	100m	3.64E-09		
	Inside	9.38E-10		
PHFSE	100m	5.63E-08		
	Inside	7.34E-09		
ME	100m	1.86E-07		
	Inside	0.00E + 00		
SE	100m	0.00E + 00		
	Inside	1.06E-07		
LOPC	100m	7.73E-08		
	Inside	0.00E + 00		
LOEP*	100m			
	Inside			
EE	100m	1.09E-08		
	Inside	7.20E-08		
Total =	100m	8.93E-07	1.00E-06	1.00E-04
Total =	Inside	1.30E-05	1.00E-06	1.00E-04

* Note: In-vessel LOCA events are incorporated into LOVA and LOPC. LOEP events are incorporated into LOFA and LOPC.

9. REVIEW OF SAFETY CASE AND APPROVAL OF TRITIUM COMMISSIONING

The documents produced at the various stages of the safety case were independently peer reviewed and revised where necessary to take into account recommendations made by the peer reviewers. Peer review recommendations also included changes to management procedures and in some limited cases, plant hardware. An action list of unresolved issues was set down in a so-called "live file" which was managed by JET but with an identical copy held by the representative of the UKAEA Director of Safety. Over 700 actions were tracked in this way.

The safety case documents and the resolution of the live file actions were presented to the JET Fusion Safety Committee to ultimately seek endorsement of an Authority to Operate (ATO) for each of the two plants. The composition of the committee included a representative of the UKAEA Director of Safety and a number of experts from other tritium facilities and nuclear establishments. The conditions for proceeding for each stage in commissioning were set down in the ATOs and included reference to the plant limits and safety systems essential to meet the safety case requirements.

A final prerequisite for tritium commissioning of both AGHS and torus was the satisfactory performance in external audits which examined the management procedures.

10. CONCLUSIONS

i) A safety analysis of the JET machine and ancillary systems in the configuration for D-T operation has been carried out in accordance with modern standards as defined in the NII Safety Assessment Principles.

ii) This has shown that the radiological risks to JET workers and members of the public have been reduced to as low as reasonably practicable.

iii) The analysis confirms that off-site countermeasures are not required for plant-initiated events.

iv) The importance of the AGHS as safety system and segregated tritium processing plant has been highlighted for a number of accident scenarios.

v) The period at risk for many accident sequences is during pulsing. This is in contrast to nuclear facilities in which the risk may be essentially time-independent and is a positive safety feature as it requires explicit operator action, with numerous countdown checks to initiate a pulse.

11. ACKNOWLEDGEMENTS

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