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# **Recent Developments in Environmental Aspects** of D-<sup>3</sup>He Fueled Fusion Devices

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### Abstract

The stress on fusion safety has stimulated worldwide research in the late 1980s for fuel cycles other than D-T. With advanced cycles, such as D-D, D-<sup>3</sup>He, p-<sup>11</sup>B, and <sup>3</sup>He-<sup>3</sup>He, it is not necessary to breed and fuel large amounts of tritium. The D-<sup>3</sup>He fuel cycle in particular is not completely aneutronic due to the side D-D reactions. Neutron wall loadings, however, can be kept low (by orders of magnitude) compared to D-T fueled plants with the same output power, eliminating the need for replacing the first wall and shielding components during the entire plant lifetime. Other attractive safety characteristics include low activity and decay heat levels, low-level waste, and low releasable radioactive inventory from credible accidents.

There is a growing international effort to alleviate the environmental impact of fusion and to support the most recent trend in radwaste management that suggests replacing the geological disposal option with more environmentally attractive scenarios, such as recycling and clearance. We took the initiative to apply these approaches to existing D-<sup>3</sup>He conceptual designs: the ARIES-III power plant and the Candor experiment. Furthermore, a comparison between the radiological aspect of the D-<sup>3</sup>He and D-T fuel cycles was assessed and showed notable differences. This report documents the comparative assessment and supports the compelling safety argument in favor of the D-<sup>3</sup>He fuel cycle.

### **1. Introduction**

Nuclear fusion is seen as a "clean" source of energy. However, the attractive safety and environmental potential of fusion can only be fully realized by a design in which more attention is paid to reducing the impact of materials activation and tritium inventory [1]. Activated materials, generated by neutron interactions with the plant structure, will be removed from the plant during routine component replacements, and finally decommissioned at the end-of-life. Tritium is a mobile and soluble radionuclide with safety-related handling concerns.

Demonstration of "Passive Safety" (without active safety systems to control and mitigate the consequences of the worst reasonably conceivable accidental sequences) is a key issue for showing a clear advantage of fusion, in view of its public acceptance. Several studies in the U.S. [2] and Europe [3] intended to assess the fusion safety and also evaluated the power plant behavior in case of the worst reasonably conceivable accidents.

Since the main source of fusion waste (70-90%) is the relatively low-level activated materials (well shielded from the plasma) it is appropriate to explore options that minimize the use of existing geological repositories or perhaps eliminate the need for building new ones [4]. For this purpose, the fusion development path and waste management strategy should aim at:

- Declassification of the slightly activated materials to non-active materials (clearance), based on the most recently issued clearance guidelines [5,6].
- Recycling the moderately radioactive materials within the nuclear industry.
- Employing aneutronic fuel cycles.

If all the materials can be cleared after a relatively short interim storage, then fusion may satisfy the so-called "zero-waste" option [7]. For deuterium-tritium (D-T) fueled devices, it has been found that, with appropriate materials selection and design, an efficient strategy for waste minimization is applicable [8]. In particular, for ARIES [2,9] and PPCS [3,10], most of the materials can be cleared, while the rest could theoretically be

eligible for recycling within the nuclear industry. However, even if feasible in theory, the zero-waste option will be difficult to achieve; a certain amount of radioactive materials from decommissioning – even if recyclable – may need to be disposed of as radioactive waste. The production of such activated materials cannot be avoided in fusion power plants by means of material choices; it always exists if a D-T fuel cycle is used. Studies [10,11] have shown that it is practically impossible to reduce the long-term radioactivity of those materials to levels allowing clearance. A further step is necessary if the passive safety and zero-waste goals are to be achieved.

#### 2. Deuterium-Helium-3 Fuel Cycle

Most of the studies and experiments on nuclear fusion are currently devoted to the D-T fuel cycle, since it is the easiest way to reach ignition. The stress on fusion safety has stimulated worldwide research on fuel cycles other than DT, based on 'advanced' reactions, such as deuterium-deuterium (D-D) and deuterium-helium-3 (D-<sup>3</sup>He). With these cycles, it is not necessary to breed large amounts of tritium for plasma fueling. The D-D fuel cycle produces much more neutrons compared to the D-<sup>3</sup>He cycle because of the direct production of tritium. The D-<sup>3</sup>He cycle is not completely aneutronic. It has a very low presence of energetic fusion neutrons, due to side D-D reactions generating 2.45 MeV neutrons and T and the side D-T reactions generating 14.1 MeV neutrons.

D-<sup>3</sup>He fusion has its own set of issues and concerns, such as the availability of <sup>3</sup>He [12] and the attainment of the higher plasma parameters that are required for burning [13]. However, it offers advantages, such as the possibility to obtain electrical power by direct energy conversion of the charged particles [14,15,16]. Fusion power based on D-<sup>3</sup>He plasmas would not need a blanket to breed tritium, and also it may avoid the need to produce electrical power indirectly, via the usual heating of a thermo vector fluid (such as water or liquid metal) that is used to drive a thermodynamic cycle with turbines [17]. In other words, D-<sup>3</sup>He fueled fusion power plants employing these technologies have no similarity with D-T fueled fusion or fission plants. References 18 and 19 address the pertinent issues of utilizing today's technology and the strategy for D-<sup>3</sup>He fusion development.

Several studies have addressed the physics and engineering issues of D-<sup>3</sup>He fueled power plants [20]. High beta and high field innovative confinement concepts, such as the field-reversed configuration [21,22], Ring Trap concept [23], and, to a lesser extent, the tokamaks are suitable devices for advanced fuel cycles. In the late 1980s and early 1990s, the University of Wisconsin Fusion Technology Institute developed a series of tokamak-based D-<sup>3</sup>He Apollo designs [24-28] while the D-<sup>3</sup>He ARIES-III tokamak was developed within the framework of the ARIES project [29-31]. The Apollo series, along with ARIES-III and other studies, demonstrated the reduction in radioactivity and the particularly advantageous effect of the D-<sup>3</sup>He safety characteristics [32-35] that include low activity and decay heat levels, low-level waste (WDR < 0.1), and low releasable radioactive inventory from credible accidents. Due to the lack of data, no clearance indices were evaluated in the early 1990s.

The Alcator class of compact, high-field experiments were the first to be proposed in order to achieve fusion ignition conditions on the basis of existing technology and the known properties of high-density plasmas. Good confinement and high purity plasmas have been obtained by high field machines, such as Alcator/Alcator C/Alcator C-MOD at the Massachusetts Institute of Technology [36] and Frascati Torus Upgrade (FT/FTU) at ENEA in Italy [37]. Ignitor is a compact high magnetic field tokamak, aimed at reaching ignition in D-T plasmas for periods of a few seconds [38-40]. A design evolution of Ignitor in the direction of a reactor using a D-<sup>3</sup>He fuel cycle has been proposed. A feasibility study of a high-field D-<sup>3</sup>He experiment of larger dimensions and higher fusion power than Ignitor, but based on the Ignitor core technologies, has led to the proposal of the Candor fusion experiment [41,42].

It is of interest to compare the safety and environmental features of the D-<sup>3</sup>He and D-T fueled power plants (ARIES-III vs. ARIES-CS) and experimental devices (Candor vs. Ignitor) and highlight the clear advantages of the D-<sup>3</sup>He system. The following sections cover the comparative analyses.

## 3. Power Plants: ARIES-III (D-<sup>3</sup>He) and ARIES-CS (D-T)

One of the significant impacts of the D-<sup>3</sup>He cycle is the low neutron wall loading (NWL), softer neutron spectrum, and low radiation damage that allow the first wall (FW) to survive the entire 40 full power years (FPY) of plant operation without a need for replacement. The permanent FW and shielding components not only increase the availability of the plant, but more importantly, reduce the radwaste stream.

The D-<sup>3</sup>He neutrons carry a small fraction of the fusion power (4-5%). About 30% of the neutrons are at 14 MeV and 70% at 2.45 MeV. Nevertheless, the 14.1 MeV neutrons carry most of the energy (70%) to the FW and produce most of the damage. The average wall loading is relatively low (~0.1 MW/m<sup>2</sup>) compared to D-T fueled plants, but the plasma surrounding components service ~40 FPY, meaning 4 MWy/m<sup>2</sup> fluence. This compares to ~15 MWy/m<sup>2</sup> for a D-T system having a limited service lifetime of 4-5 FPY for the FW/blanket. Besides the neutron spectral differences, the factor of four reduction in activity alleviates the activation problems but the D-<sup>3</sup>He system cannot reach the "zero-waste" status yet.

ARIES-III [29] generates 1000 MW net electric power and relies on thermal conversion of fusion energy to electricity. An isometric view of ARIES-III is shown in Fig. 1. The shield and vacuum vessel (VV) protect the superconducting magnet for plant life (40 FPY). The structural material of choice is the modified HT-9 ferritic steel (FS). The outboard 70 cm thick shield and 10 cm thick VV (~75% FS and ~25% organic coolant) occupy approximately half the space needed for the D-T cycle. The 40 FPY and 85% availability have been represented as 0.85 y irradiation period followed by 0.15 y downtime, and repeated for 47 y. The ALARA activation code [43], DANTSYS transport code [44], and FENDL-2 data library [45] have been used throughout the study.



Fig. 1. Isometric view of ARIES-III.



Fig. 2. Specific activity of ARIES-III components. Only the bioshield activity of the innermost segment is displayed.

As a source term, the variation of ARIES-III activity with time after shutdown, shown in Fig. 2, has been used to evaluate the radiological hazards of the individual components. The results reported herein pertain to the fully compacted, 100% dense solids only. No attempt has been made to assess the activation of the coolant.

Since its inception, the ARIES-III design [29] focused on the disposal of all active materials. We evaluated the waste disposal rating (WDR) for a fully compacted waste using the most conservative waste disposal limits developed by Fetter [46] and NRC-10CFR61 [47]. For individual components, the WDR (ratio of the specific activity at 100 y after shutdown to the allowable limit summed over all radioisotopes) is less than one, meaning all components qualify as Class C LLW. The WDRs of the shield, VV and bioshield (not shown in Fig. 1) are very low (< 0.1), to the extent that these components could qualify as Class A low-level waste (LLW), the least hazardous waste according to the U.S. waste classification guidelines. Excluding the bioshield, ~20% of the waste (mainly the Incalloy coil case and winding pack of the magnet) is Class C. The remaining ~80% (shield and VV) would fall under the Class A LLW category.

The recent introduction of the clearance category for slightly radioactive materials and the development of radiation-hardened remote handling equipment opened the possibility to recycle and clear the majority of the ARIES-III radwaste. Scenarios for fusion radwaste management should not mandate geological disposal in repositories, but must also consider recycling and reuse within the nuclear industry, and/or clearance or release to the commercial market if the materials contain traces of radioactivity [4,9,10,48-50]. Recycling and clearance can be regarded as an effective means to reduce the fusion radwaste stream. The reason is that clearable materials will not be categorized as waste and the majority of the remaining non-clearable materials can potentially be recycled indefinitely and therefore, will not be assigned for geological disposal.

Clearance guidelines have been recently issued by the U.S. Nuclear Regulatory Commission (NRC) [5], International Atomic Energy Agency (IAEA) [6], and other organizations [50]. Based on a detailed technical study, the U.S. 2003 NUREG-1640 document [5] contains estimates of the total effective dose equivalent (from which the clearance index can be derived) for 115 radionuclides that could be present in activated steel, copper, aluminum, and concrete from decommissioning of nuclear facilities. The NRC has not yet issued an official policy on the unconditional release of specific materials. Herein, the proposed annual doses reported in the NUREG-1640 document will be referred to as the proposed U.S. limits. A clearance index (CI) can be computed as the weighted sum of all nuclide specific activities (in Bq/g) divided by the corresponding clearance limits.

Figure 3 indicates that all ARIES-III in-vessel components (shield, VV, and magnet) cannot be cleared from regulatory control even after an extended storage of 100 y, according to the IAEA clearance guidelines [6]. This statement holds true as well for the proposed U.S. clearance standards [5]. The bioshield qualifies for clearance, however. Representing 85% of the total volume, the clearable bioshield saves a substantial disposal cost and, more importantly, free ample space in the repositories for other radwaste. Since the ultimate goal is to separate the constituents of the component for recycling and reuse by industry, the ARIES approach for handling the cleared components (CI < 1) is to reevaluate the CIs for the constituents [9,49]. The entire component could have a CI < 1, but the individual constituents may not and vice versa, requiring further segregation of the active materials based on constituents rather than components.

We propose another approach to deal with sizable components, such as the 2 m thick bioshield. It should be segmented and reexamined. As such, the bioshield was divided into four segments (0.5 m each) and the CIs reevaluated for the constituents (85% concrete and 15% mild steel, by volume). The results indicate that the innermost segment has the highest CI while the outer three segments meet the clearance limit within a few days after decommissioning. As Figs. 4 and 5 indicate, the mild steel is a major contributor to the CI although its volume fraction is only 15%. It can be cleared in 4-10 y while the concrete requires a shorter storage period of only one year.



Fig. 3. Decrease of ARIES-III clearance index with time after decommissioning. Only the bioshield CI of the innermost segment is displayed.



Fig. 4. Variation of CI with time for steel contained in innermost segment of bioshield.



Fig. 5. Variation of CI with time for concrete contained in innermost segment of bioshield.



Fig. 6. Sensitivity of dose to decay time after operation.

As an alternate approach, we applied the recycling scenario to the non-clearable in-vessel components (shield, VV, and magnet). The variation with time of the recycling dose shows a strong material dependence (refer to Fig. 6). All components can potentially be recycled using conventional and advanced remote handling (RH) equipment. Applying the As-Low-As-Reasonably-Achievable principle, the 1 micro Sv/h hands-on design limit considered throughout the ARIES studies is a factor of 10 below the absolute limit of 10 micro Sv/h. This limit is currently under revision by the international safety community. The potential change to the final results is minimal if the higher new limit remains within a factor of 2-3. <sup>54</sup>Mn (from Fe) is the main contributor to the dose of the shield and VV. Storing the shield temporarily for several years helps drop the dose by a few orders of magnitude before recycling. After several life cycles, advanced RH equipment could recycle all components shortly after decommissioning.

To make a comparison with D-T systems, we selected the most recent ARIES design developed by the ARIES team: the compact stellarator [51,52]. It is a 1000 MW<sub>e</sub> design that combines advanced physics and engineering approaches to minimize the major radius and the overall size. The design is compact and generates less radioactive waste relative to predecessor stellarator designs. Table 1 summarizes the key features of ARIES-III and ARIES-CS along with the WDR of the in-vessel components. Both designs employ Nb<sub>3</sub>Sn as the magnet superconductor, resulting in a Class C LLW. The dominant Class A WDR of ARIES-III components represents a clear advantage for D-<sup>3</sup>He over DT.

Figure 7 compares the activity of both designs. Even though the D-T blanket service lifetime is relatively short, it generates higher activity than the 40 FPY D-<sup>3</sup>He shield. A similar observation can be made for the VV despite the identical service lifetime. Both designs would require advanced RH equipment to recycle all components [4], but the cooling period is quite shorter for ARIES-III. The radioactive inventory represents another interesting metric. The breakdown of the fully compacted waste is displayed in Fig. 8. The ARIES-CS blanket volume reflects the nine replacements required during operation while none is needed for ARIES-III. Note the remarkable difference between

the total radwaste volume generated by ARIES-III and ARIES-CS. Future studies should include in this particular comparison the volumes of D-T tokamaks that typically generate less radwaste than stellarators.

	ARIES-III	ARIES-CS
Fuel cycle	D- <sup>3</sup> He	D-T
Major radius (m)	7.5	7.75
Minor radius (m)	2.5	1.7
Average NWL (MW/m <sup>2</sup> )	0.1	2.6
Structure	MHT-9	MF82H
Coolant	OC	He/LiPb
Breeder		LiPb
Thermal efficiency	44%	42%
Blanket WDR		Class C
Shield WDR	Class A	Class C
VV WDR	Class A	Class A
Magnet WDR	Class C	Class C

Table 1. Key parameters for ARIES-III and ARIES-CS designs



Fig. 7. Comparison of ARIES-III and ARIES-CS specific activities.



Fig. 8. Comparison of ARIES-III and ARIES-CS waste inventory.

## 4. Experimental Devices: Ignitor (D-T) and Candor (D-<sup>3</sup>He)

Ignitor is a proposed compact high magnetic field tokamak, aimed at reaching ignition in D-T plasmas for short periods of a few seconds [38-40]. Pulses at different power levels are planned, with either D-D or D-T operation, distributed over a global operation time of 10 calendar years. Ignitor main safety requirements are:

- The experiment must protect the health and safety of the facility personnel and public, by maintaining an effective defense against hazards.
- Ignitor must maintain an operation that is environmentally acceptable to present and future generations and to satisfy the two basic requirements for environmental feasibility of fusion:

- 1. No need for public evacuation in the case of worst accident scenario.
- No production of waste that could be a burden for future generations, i.e., minimization of the production of long-lived radwaste.

Finally, the experiment must be easily sited in Italy, according to both international and country regulations. The European Union has recently studied a waste management strategy for fusion radioactive materials [3,10] based upon recycling and clearance. We have applied this strategy to Ignitor, using the latest set of IAEA clearance limits [6].

We have always assumed that adequate detritiation of material is carried out before disposal. The total tritium inventory in Ignitor is limited to a few grams. Results of the application of this strategy to Ignitor are shown in Table 2. The Inconel-625 vacuum vessel, Cu magnets, and Mo first wall could be easily recycled within the nuclear industry, while other materials (part of the C-clamp and the cryostat) could be cleared, i.e., declassified to non-active materials. This indicates that Ignitor does not completely fulfil the requirement of no production of waste, but minimizes it to a few cubic meters that could be recycled within the nuclear industry. Note that for the Ignitor and Candor, experimental machines, the 10 micro Sv/h hands-on recycling limit has been adopted. It is computed by dividing the annual dose limit for exposed workers (20 mSv/y) by the standard 2000 hours/year exposure. For ARIES power plants, a ten-fold reduced limit of 1 micro Sv/h has been considered applying the As-Low-As-Reasonably-Achievable principle.

Concerning accident analyses, results obtained elsewhere for Ignitor show that, in case of the worst case DBA (Design Basis Accident), dose to the most exposed individuals (MEI) is around 1 mSv [53]. Most of the radioactive releases deal with tritiated materials and activated materials from the magnet components.

Component	Material	Classification	Necessary	Volume
			Decay Time	(m <sup>3</sup> )
Vessel	Inconel-625 alloy	Recycling or LLW	60 years	4.4
First Wall	Molybdenum	Hands-on recycling	10 years	2
Magnet	Copper	Hands-on Recycling	60 years	12.2
C-Clamp Structure	AISI 316 steel	Hands-on recycling (40%) Clearance (60%)	40 years	24
Cryostat	Composite material	Clearance	20 years	1.1

Table 2. Classification of Ignitor radioactive materials and components

Ignitor is an experimental machine, not a power plant, and uses D-T plasmas. However, the plasma density limit in Ignitor is well above the optimal density for D-T ignition; it is potentially suitable for the higher densities required for D-<sup>3</sup>He burning. In fact, Ignitor has also been designed to explore conditions where 14.7 MeV protons and 3.6 MeV alpha particles produced by the D-<sup>3</sup>He reactions can supply a significant amount of thermal energy to a well-confined plasma [42]. A design evolution of Ignitor in the direction of a reactor using a D-<sup>3</sup>He fuel cycle has been proposed. A feasibility study of a high-field D-<sup>3</sup>He experiment with larger dimensions and higher fusion power than Ignitor, still based on the Ignitor core technologies, has led to the proposal of the Candor fusion experiment [41,42,54]. The main characteristics of the Candor machine are the following: the major radius R<sub>o</sub> is about double that of Ignitor, plasma currents up to 25 MA with toroidal magnetic fields B<sub>T</sub> of 13 T can be produced. Unlike Ignitor, Candor would operate with higher values of poloidal beta around unity and the central part of the plasma column in the second stability regime of the major plasma instabilities. To produce a similar strong magnetic field in a larger machine, the toroidal field coils are divided into two sets of coils and the central solenoid (air core transformer) is placed between them in the inboard side.

In Candor, the D-<sup>3</sup>He burning regime can be reached by a combination of ICRF heating and alpha particle heating due to D-T fusion reactions that take the role of a trigger. Thanks to this fact, and unlike other proposed D-<sup>3</sup>He fusion experiments, Candor is capable of reaching D-<sup>3</sup>He ignition on the basis of existing technologies and knowledge of plasma confinement. With the initial use of D-T, where tritium comprises some 50% of an initial lower density D-T plasma, but is not added thereafter, the need for intense auxiliary heating, which is one of the main technological drawbacks of D-<sup>3</sup>He ignition, would be considerably alleviated, becoming feasible with the present technology. However, this method has the disadvantage of having a higher neutron flux (due to D-T reactions) than "pure" D-<sup>3</sup>He plasmas, generating a neutron flux transient when passing from the initial DT trigger reaction to the final D-<sup>3</sup>He burning plasma. The characteristic times over which the Candor plasma discharge can be sustained are longer by more than a factor of 4 than those of Ignitor. An illustration of the Candor experiment is shown in Fig. 9.



Fig. 9. Illustration of the Candor experiment.



CANDOR Activation at 10 y, 30 y, 50 y after shutdown

Fig. 10. Candor activation.

Tritium inventory in Candor is expected to be very small and should not present a problem from the safety viewpoint. The neutron-induced activation of Candor is quite moderate. Figure 10 shows the activity of the various components.

To minimize the quantity of active materials that requires long-term storage, maximum use should be made of both recycling within the nuclear industry and clearance. Examining the results for average materials/components displayed in Fig. 11, we conclude that hands-on recycling is possible within less than 10 years of decay for SS316 and in less than 30 years for copper.

Concerning the management of Candor radioactive waste outside the nuclear industry, all components can be declassified as non-radioactive materials ("cleared") within cooling times varying from 50 to 100 y, according to the IAEA clearance guidelines. The results for the average materials/components are given in Fig. 12. Clearance is possible within less than 50 years of decay for SS316 and in about 100 years for copper.



Fig. 11. Dose rates for Candor materials (average).

The results for the two experimental machines (Ignitor and Candor) are compared in Fig. 13. It turns out that Candor results are more environmentally attractive than those obtained for Ignitor. Candor appears to be a step forward toward the design of a "clean" fusion energy machine.



Fig. 12. Clearance index for Candor materials (average).



Fig. 13. Amount of recyclable and clearable materials in Ignitor and Candor.

#### 5. Conclusions

This paper focuses on the safety and radioactive waste issue for fusion. Innovative solutions in those areas could be a clear advantage of fusion in view of its ultimate safety and public acceptance. Concerning the waste, recycling and/or clearance (i.e., declassification to non-radioactive materials) of all components, after a sufficient period of interim decay, should be the goal for an environmentally attractive fusion plant. Demonstration of passive safety would also be necessary. This translates into negligible doses to the public even in the case of the worst conceivable accidents with radioactive environmental release.

As a further step towards the waste minimization and passive safety goals, the features of fusion devices based on alternative advanced fuel cycles have been examined. In particular, the advanced D-<sup>3</sup>He fuel cycle offers outstanding environmental advantages, such as the quite low presence of tritium, neutrons, and activated materials. Ignition of D-<sup>3</sup>He plasmas, however, is more difficult to achieve compared to D-T plasmas.

For a representative D-<sup>3</sup>He power plant (ARIES-III), we estimated the highest possible activity to evaluate the disposal, recycling, and clearance options for managing the radwaste after decommissioning. We compared the results to a D-T system to highlight the differences and the environmental impact. The results show that all ARIES-III invessel components qualify as Class A waste, the least hazardous type based on the U.S. guidelines. Potentially, all components can be recycled using conventional and advanced remote handling equipment. The bioshield contains traces of radioactivity and can be cleared from regulatory control after a relatively short period of time (~10 y).

The zero-waste goal for fusion power plants using either D-T or  $D^{-3}He$  fuel cycle will be difficult to obtain. A specific amount of radioactive materials from decommissioning – even if all components can be recycled within the nuclear industry – should be disposed of as radioactive waste. Most likely, these materials will meet the requirements for classification as low-level waste.

Results obtained for the  $D^{-3}$ He Candor experiment show that no environmental problems arise from such a device, from the radiological point of view, even with the presence of D-T plasma triggering. Candor does reach the zero-waste option as all wastes can be cleared within 100 y.

The D-<sup>3</sup>He cycle offers safety advantages and could be the ultimate response to the environmental requirements for future nuclear power plants. Furthermore, the low neutron production helps overcome some of the engineering and material hurdles to fusion development. Studies for the development of advanced fuel cycles should be carried out in parallel with the current mainstream fusion pathway that primarily focuses on D-T tokamaks, such as ITER, test facilities, Demo, and power plants.

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