Fusion Safety, Environmental and In-Vessel Tritium Opportunities Afforded by ITER

David Petti
Fusion Safety Program
Idaho National Engineering and Environmental Laboratory

Abstract

The decision of the US to re-enter the ITER project affords the US fusion program some important opportunities in the areas of fusion safety, environment and in-vessel tritium research. In this white paper, the specific activities that the US could be engaged in are outlined and the benefit to the US and the ITER program are defined.

1. Historic background

The US fusion safety program, established in 1979, has spent the past 25 years establishing the regulatory requirements appropriate for a fusion power plant, performing the requisite safety and environmental R&D and developing the computer codes that would be needed for safety analysis of a fusion power plant. All of this work is needed for a fusion machine like ITER to obtain regulatory approval for construction and operation. As such, the US had a very strong presence in the ITER Engineering Design Activity (EDA). We led the development of much of the seminal thinking related to regulatory approach and the integration of safety in the design, performed much of the most difficult parts of the ITER safety analysis, and produced much of the required critical R&D needed to support the safety analysis. The following sections outline the specific tasks where the US has the requisite expertise to help the ITER project and would strategically improve the US base program for the longer term.

2. Regulatory support and safety/design integration

Precedent is very important in regulatory matters. The specific regulatory requirements imposed by the host on ITER can have ramifications for ANY future fusion facility built anywhere in the world. The US established fusion-specific safety requirements and the corresponding technical rational in the development of the US DOE Fusion Safety Standards [1,2,3]. We used these powerful technical arguments to help influence the nature of the requirements during the ITER EDA. Since our absence some recidivism has occurred especially in terms of implementation of safety requirements in the design. We need to be actively engaged in discussions with the ITER safety team and the regulatory authorities of the potential hosts to "push back" on requirements that do not make sense for fusion. Our history in the EDA suggests that we can still influence decision-makers with such arguments and given the stakes involved the potential is well worth the modest investment.

The US goal has always been to obtain the maximum operational flexibility for the machine, given its experimental nature, yet still meet a set of regulatory requirements that provide adequate safety and make sense for the technology. When ITER enters the operation phase, limits may be placed on operation based on the traditional fission approach used in the host countries. We need to be involved to ensure that we can retain the maximum flexibility and not hinder future operations. Beyond regulatory support, safety-design integration is an area where the US has historically been quite strong. It is a continuous effort, which requires a good knowledge of the safety requirements, the ITER safety approach, the safety analysis results and the on-going design. This is how the safety requirements are integrated into the design of the facility.

3. Safety analysis and safety computer code verification and validation

Much of the ITER FEAT safety report is based on the report developed for the ITER EDA design. Many of the calculations supporting the ITER EDA safety report were performed by the US with state-of-the-art fission computer codes (e.g., MELCOR) that were converted by INEEL for use in fusion. In the course of the production of the general site-specific safety report (without direct support of the US) the ITER team confirmed the importance of the US MELCOR code [4]. This is the main tool for comprehensive safety analysis of the tokamak, its cooling system, and the rooms around the tokamak. The strength of MELCOR is its unique integration of fluid-dynamic description of the cooling loop itself, the containment volumes and the transport of radioactivity. Various essential fusion-specific features were included in MELCOR by the US during the ITER-EDA phase [5], which are the pre-condition for a meaningful application for ITER. These include the description of cryogenic structures in the cryostat, the extension of water properties to below freezing temperatures, proper treatment of quasi-2d heat transfer in near plasma structures and the tracking of tritium transport. Given the central role of this code, the INEEL fusion-specific version should be merged with the most up-to-date version of MELCOR and aim for a new baseline release that could form the basis for calculations requested during the process of obtaining the construction license for ITER. In addition, application of MELCOR for a limited set of key accident sequences would be useful for verification efforts to enhance the quality of ITER safety reports.

Greater US participation in validation studies of the safety codes used in ITER is also necessary. All regulatory bodies expect the codes used in the safety analysis to undergo a process to verify and validate their results. As a result, an international thermal hydraulic code benchmark and experiment validation program has been developed under IEA auspices but US involvement has been limited because of budget constraints. This is an area that is in the long-term interest of the US because such benchmarking of the codes will help support future fusion designs.

The ITER safety team has also indicated that they need more help in the area of magnet arcs. At the end of its ITER participation the US/INEEL had contributed with the first consistent safety analysis of an unmitigated quench in a TF coil [6]. This analysis was unique in the sense that the behavior of arcs, electric circuits, and molten material had to be treated in a consistent way, which has no parallel in any fission application. Application of this code and techniques to the current ITER design for all 3 types of coils (TF, PF and CS) is needed. In addition extension of the analysis to external coils arcs would be very useful since up to now only very simplified (and therefore conservative) models exist. Modest investment in this entire area would yield very useful results.

4. Tokamak dust source term

Dust characterization is important for ITER since dust is one of the two principal radioactive source terms in ITER (the other is tritium). The US/INEEL has performed a comprehensive R&D program to survey the dust size distribution expected in tokamaks (for example, [7, 8]). The conclusion from this extended R&D program is the current basis of the ITER safety reports. However, further R&D efforts are needed to have a more robust conclusion on the distribution of very small particles (<0.5 μm). This would involve new approaches in measurement techniques including the use of electron microscopes instead of optical methods and modeling of the formation of dust in tokamaks. This R&D on dust formation modeling and characterization from existing machines may help ease the limits currently imposed on ITER

The ITER strategy with dust is to set administrative limits on the amount of dust allowed inside the ITER vacuum vessel (VV). However, ITER has not yet developed and demonstrated a method for monitoring (needed to demonstrate that the administrative limits are respected) and removing dust (needed to continue operation should the administrative limit be exceeded). An R&D program is needed with a small experiment to measure dust mobilization in ITER relevant condition (air ingress into VV), and to test various dust measurement and removal techniques. This should help better validate proposed confinement strategies and estimates of routine dust releases during maintenance activities in the tokamak.

5. In-vessel tritium source term

The in-vessel tritium inventory is the second major component of the ITER source term. The influence of mixed materials in the vacuum vessel (C, Be, and W) on the inventory has been studied, but many questions remain unanswered. For example, the influence of erosion of Be and C during plasma transients (ELMs, disruptions, VDEs) on the tritium co-deposition rate is still uncertain. Furthermore, the research to date indicates that the use of carbon even in a limited way as a plasma-facing component inside the vacuum vessel will result in the buildup of large inventories of tritium without detritiation. Because the in-vessel inventory is a key safety parameter, validation of all the assumptions used in the development of the inventory estimate, including computer code calculations (e.g., TMAP), used in the safety analysis is critical. Additional focused R&D will be needed to reduce some of the remaining uncertainties.

Furthermore, the current ITER strategy with respect to remaining below the administrative tritium inventory limit will need to be evaluated. The goal is to allow the greatest flexibility of operation given the current limits and methods for removing tritium from the plasma facing components. The Tritium Plasma Experiment will be used to study some of the fundamental behavior under the most prototypic plasma conditions and to validate the inventory control strategy including experiments on tritium uptake, erosion phenomena, and advanced in-vessel detritiation techniques. The US has unique capability in this area as evidenced by the recent request from ITER for the US to continue mixed materials research. The additional research and access to studying such behavior in a large-scale tokamak will give us unprecedented insight into plasma materials interactions under prototypic conditions and allow us to improve future machine designs. Better estimates of the tritium inventory will allow greater flexibility with respect to establishing the margin to the in-vessel inventory limit, and will permit more accurate assessments or radiological risk to workers during maintenance and the public under off-normal conditions.

6. Verification activities during operation

There will be many assumptions used in the ITER safety analysis that can only be verified during the early machine operation and commissioning activities. In other cases, operation at full power will be required to obtain the requisite data (e.g., tritium inventory in key components, activation product inventories, decay heat, operational behavior of pressure suppression system, tritium cleanup and dust removal systems). The critical assumptions will probably be outlined during the regulatory review and approval process and thus the verification activities performed during early operation will be critical to obtaining regulatory approval for the next step in operations. (We assume here that the ITER team will use a staged approach to regulatory approval based on the successful approach used for other large nuclear technology projects in the US and around the world.) Such an activity would have usefulness for the design of future fusion machines as conservatisms in the safety approach could be relaxed as real world data on an actual large-scale tokamak are gathered, assessed and understood.

7. Conclusions

The decision of the US to re-enter the ITER project affords the US fusion program some important opportunities in the areas of fusion safety, environment and in-vessel tritium research. In this white paper, the specific activities that the US could be engaged in include:

- Regulatory support and safety/design integration
- Safety analysis and safety computer code verification and validation
- Tokamak dust source term
- In-vessel tritium source term, and
- Verification activities during operation

The benefits to the US and the ITER program were also defined.

8. References

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