4. SAFETY AND LICENSING

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4. SAFETY AND LICENSING

4.1. INTRODUCTION

During the Assessment Phase the safety and licensing activity focused on four areas: (1) a review of Nuclear Regulatory Commission (NRC) documentation as it relates to jurisdiction, regulation, licensing and tritium requirements in the context of fusion power systems [1]; (2) the evolution of a potential pathway for development of fusion regulatory policy and requirements [2]; (3) the establishment of top-level safety and licensing requirements for fusion power systems [3]; and (4) the development of a methodology to demonstrate that sheltering and evacuation is not required for fusion power systems [4]. In addition to the four activities noted above, several members of the safety and licensing activity interfaced with, and contributed to, the DOE Fusion Safety Standard Activity.

4.2. REVIEW OF NRC DOCUMENTATION

The review of NRC documentation was undertaken in order to anticipate the role that the NRC might have with regard to fusion power systems constructed in the United States. The results of our review are reported in four documents issued by the Project. The key findings of these documents are summarized in the following paragraphs.

The first issue investigated was NRC jurisdiction with regard to fusion power systems [1]. In this context the key findings are:

1. Fusion processes are included in the NRC scope by virtue of the Atomic Energy Act of 1954 and subsequent legislation.

2. It appears that the NRC would have jurisdiction of both Commercial and Demonstration Fusion Power Systems constructed in the United States.

Given that the NRC might have jurisdiction over fusion power systems, it seemed prudent to review NRC regulations regarding fission power plants in order to take advantage of the differences between fusion and fission power. The key findings of this review [2] are:

1. To anticipate and influence possible NRC regulations of fusion power, one must be cognizant not only of 10CFR0-199 but also the lower tier documents.

2. For fusion power plant designers working on structures, systems, and/or components that are similar or identical to those used in a fission power plant, a recommended starting point for NRC regulatory criteria is the Standard Review Plans (NRC NUREG-0800).

3. For fusion power plant designers working on structures, systems, and/or components that are unique to a fusion power plant, the Standard Review Plans may provide guidance for evolving regulatory criteria.

If one assumes that the NRC will be responsible for licensing commercial fusion power plants, it is important to understand the implications of such an eventuality. Therefore, the historical licensing process and the "one-step" licensing process used for fission power plants were reviewed. The key finding of this review [3] was that contested hearings can have an enormous impact on the time-line for licensing and every effort should be made to resolve issues prior to hearings.

Finally, since tritium will be the most prevalent radio-isotope in a fusion power plant, NRC requirements with regard to tritium were reviewed [4]. It was found that currently, very little tritium is under NRC jurisdiction, and thus, there is little regulatory record pertaining specifically to tritium. Therefore, at this time it is not possible to determine what radiological safeguards and accountability requirements the NRC would impose on tritium inventories in fusion power plants. It was also noted that present DOE tritium accountability requirements do not appear appropriate for fusion power systems.

4.3. PATHWAY FOR DEVELOPMENT OF FUSION REGULATORY POLICY AND REQUIREMENTS

Given that the NRC might have regulatory jurisdiction over fusion power plants constructed in the U.S., it appeared advisable to lay out a "strawman" pathway for developing fusion regulatory policy and requirements. This strawman pathway is shown in Table 4.3-I.

Table 4.3-I.
A Pathway for Development of Fusion Regulatory Policy & Requirements

Time frame	Action required By	Proposed action
2 years	DOE/OFE with fusion community review	Prepare a white paper on fusion characteristics & regulation, building on fusion safety standard and assuming NRC to regulate.
1 year	DOE/OFE and fusion community	Visit NRC commissioners to review white paper, obtain commissioners views as to fusion regulation & desired process.
2 years	DOE/OFE	Address issue of funding for NRC participation in development of fusion regulations.
1 year	DOE/OFE and fusion community	Based on above, develop a policy statement to be presented to the commissioners.
1 year	DOE/NRC	Conduct public meetings to review policy. obtain input, revise policy & have DOE/NRC agree to policy.
4 years	DOE/NRC	Develop draft fusion CFR requirements based on ITER experience & DEMO design & obtain comments.
4 years	DOE/NRC	Develop regulatory guidance on how to meet requirements & refine as technology evolves.
TOTAL: 15 years		

4.4. TOP-LEVEL SAFETY REQUIREMENTS

The safety and licensing activity developed top-level safety requirements for fusion power plants. First, policy statements for the public and for workers were developed. These are:

1. Operation of the facility shall not disturb the normal, day-to-day activities of the public.
2. Operation of the facility shall not expose plant personnel to risks greater than other power plants.

Next quantitative requirements for radiological, toxic and electromagnetic exposures were developed. These requirements are given in Tables 4.4-I through 4.4-III.

Table 4.4-I.
Radiological (Dose) Requirements (rem)

Description of event	Off-site level	On-site level
Normal & Anticipated Operational Occurrences	$\sim$0.01/y[(a)]	5/y[(b)]
Off-Normal Conditions and Accidents	1/Event[(b)]	5/Event[(d)]

[(a)] According to 10CFR50 Appendix 1, Acceptable Annual Dose is in range 0.003 to 0.015 rem depending on form of effluent and affected organ.

[(b)] To avoid sheltering or evacuation, dose per event cannot exceed 1 rem [5] at site boundary.

[(c)] Consistent with 10CFR20.

[(d)] Consistent with 10CFR835.1302, however, for certain activities (life-saving and protection of large populations) higher doses are allowed.

Table 4.4-II.
Toxic Material Requirements

Event	Public	Facility worker
Normal & anticipated operational occurrences	Federal, state, or local regulatory and permit requirements	Federal, state, or local regulatory and permit requirements
Off-normal conditions & accidents per event	5 times final RQs[(a)]	5 times final RQs [(a)]

[(a)] Per 40CFR302.4 and 40CFR355 App. A (RQ: Reportable quantity)

Table 4.4-III.
Static Electromagnetic Field Exposure Limit for Facility Workers[(a)]

Magnetic field	Exposure time	Body region
0.06 T[(b)]	continuous 8 h day	trunk
0.6 T[(c)]	continuous 8 h day	extremities
2.0 T[(d)]	short (a few min.)	whole Body

[(a)] Based on evaluation by Lawrence Livermore National Laboratory in 1987 [6]. Recommendations are also available for gradient and time-varying fields, but were judged to be too detailed for top-level requirements.

[(b)] Maximum average/day in peak fields > 0.5 T.

[(c)] Maximum average.

[(d)] Peak exposure limit.

4.5. METHODOLOGY FOR SAFETY ANALYSIS

A limited hazard assessment was performed on the ARIES-IV design to demonstrate the methodology for assuring design compliance with the top-level public safety requirements. The task also provided insights as to risk-sensitive areas of the ARIES-IV design.

Figure 4.5-1 outlines the methodology for evaluating the design against the top-level safety requirement. In this case the qualitative safety goal was for the plant operation to not disturb the normal day-to-day activities of the public. This in turn required that the quantitative NRC and Environmental Protection Agency (EPA) regulations for the normal and accident doses at the plant boundary be met. The accident dose is more limiting and specifically a committed effective dose equivalent of 1 rem corresponds to when sheltering is first considered [5]. To meet this off-site dose the plant must be designed to either have insufficient radioactivity that this dose level cannot be exceeded if it was released to the plant boundary or to show that the likelihood of this dose being exceeded is negligibly small (smaller than once in 5 or 10 million plant operation years). This demonstration can be by considering the design inherent resistance to radioactivity transport, by considering the reliability of engineering barriers to release, or by some combination of inherent and engineered design features. Initiating events ranging from anticipated occurrences to rare external events must be considered as well as events involving multiple independent and dependent failures of both active and passive structures, systems, and components and operator errors.

The first level evaluation is the hazard risk assessment. A range of events for the design are defined and their frequencies estimated. Best estimate release fractions of the radiologically important nuclides are determined. At this point one can judge whether the Protective Action Guides (PAGs) dose guidelines may be exceeded for the design and its sources and potential release mechanisms. If so, a more detailed evaluation including the uncertainties in the best estimate releases and doses is performed. If not, bounding lower level requirements are set on the structures, systems, and components (SSCs), *e.g.,* the radiological inventory in a given system shall not exceed a certain value. Ideally uncertainties on each of the independent variables (typically 6 to 12 counting all sources and pathways) are assessed.

The overall uncertainty in the dose for a given event is determined by utilizing a Monte Carlo code to statistically combine uncertainty distribution of each dependent variable. Dose distribution for each event can be combined, weighted by the event frequency to give the overall plant cumulative risk curve. It is the cumulative risk curve that once again is

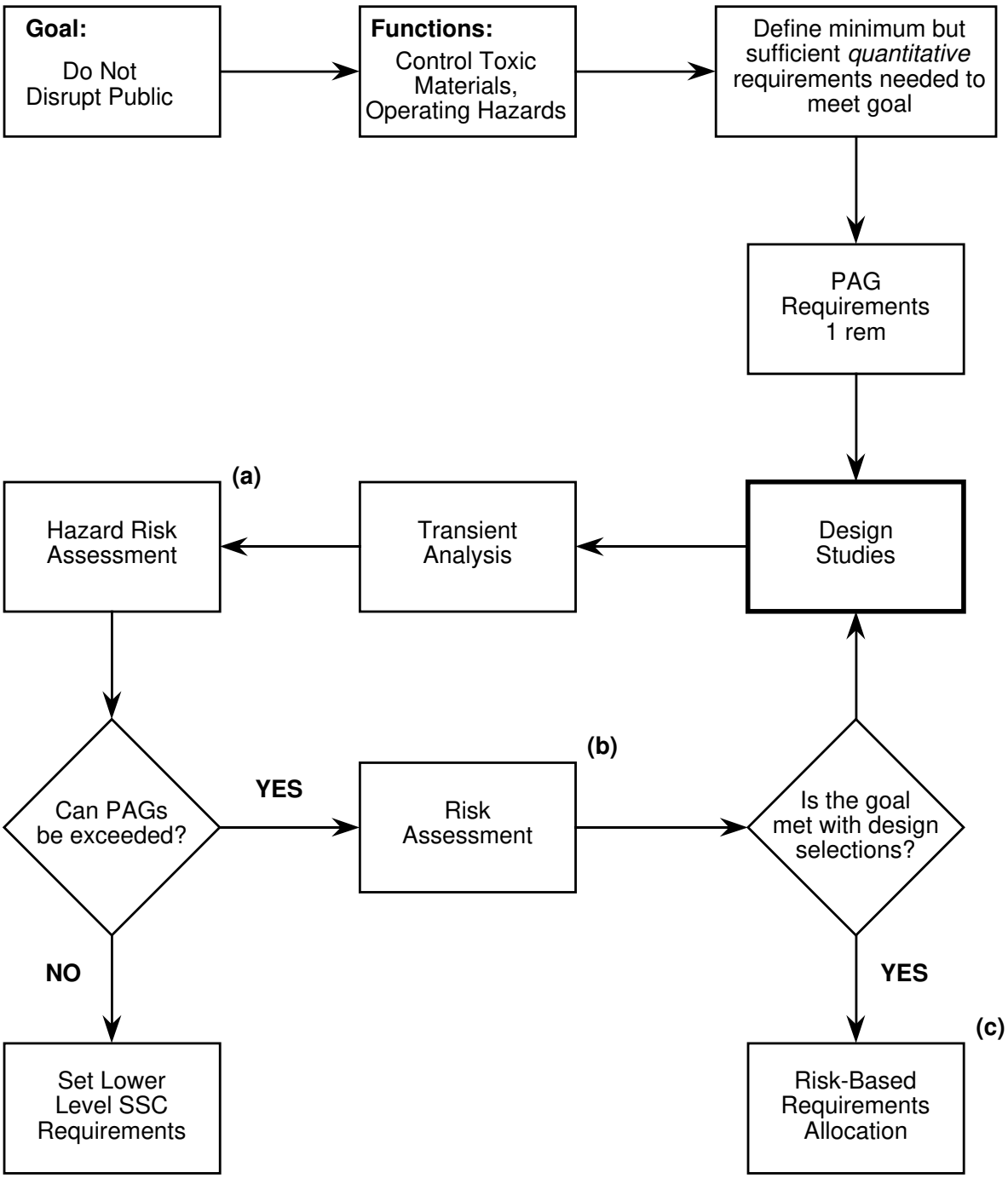

Figure 4.5-1. Logic for designing fusion facility to meet Protective Action Guidelines (PAG) requirement. The goal is to avoid "risk assessment" by meeting PAG requirements by using low-activation material and care in design.

now compared to the PAG requirement. If the goal is met, lower level risk requirements are set on the SSCs. If not, three possibilities remain to show compliance:

1. perform a still more detailed evaluation of the event frequencies and consequences,
2. specify a research program to show more favorable best estimate results and/or to narrow the uncertainty on one or more variables, such as the release fraction,
3. make design changes that lower the frequency of the high consequence events and/or lower the consequences of the high frequency events.

If the latter approach of changing the design is chosen another analysis loop is required to assure that no new high consequence events were introduced.

4.5.1. Important Events for ARIES-IV

Initiating events for existing fission plants, for advanced fission reactor designs and for fusion designs were reviewed. Representative events included primary coolant leaks, loss of off-site power, moisture ingress, and magnet failures. Engineering judgment was used to select three potential event classes thought to be important for the ARIES-IV design to meet the PAG off-site boundary doses:

1. Primary coolant leak in the blanket cooling system;
2. Primary coolant leak in a purge line;
3. Moisture ingress into the primary system.

In this limited study only the first class of events was evaluated to demonstrate the approach and methods and to provide early insights into any design issues or concerns with compliance to the top-level off-site safety requirement. Ultimately the full spectrum of events must be assessed including those with off-site releases but also those in which a design feature prevents off-site releases.

Primary coolant leak. In the ARIES-IV design the primary coolant is pressurized helium which has safety advantages (chemically and neutronically inert, single-phase). The only serious radioactivity in the helium coolant is due to other gaseous impurities in it. However, since it is pressurized it can act as a driving mechanism for transport of other radioactive sources. In the selected event, the initiating event is a failure in the blanket coolant system piping. This is followed by a plant shutdown. In this scenario the helium circulators fail immediately and are not recovered such that the blanket slowly heats up due to the decay heat and the loss of forced cooling and helium inventory.

4.5.2. Event Frequency Estimate

For this scoping study a thorough review of leakage data from high pressure systems with emphasis on nuclear power applications and specifically on gas-cooled reactors was not undertaken. Rather an estimate was taken based on an advanced light water reactor probabilistic risk assessment (PRA). From the PRA of Brown Boveri - Combustion Engineering advanced light-water reactor (ABB-CE 80) small leaks (area less than 0.05 sq. ft) were assigned a frequency of 0.003 per plant-year based on actual experience in pressurized water reactors of two primary coolant leaks in 660 reactor years of operation. Similarly, large leaks (area greater than 0.5 sq. ft) were assigned a frequency of 0.0001 per plant-year.

Given that a helium leak occurs, there is a very high likelihood (> 0.99) that the plant would be designed to reliably shutdown on low helium inventory, high building radioactivity, or other protection trips. Further, the plant would be designed to continue forced cooling. Depending on the leak size if time permits it might be prudent to initiate an intentional helium depressurization to lower the system pressure and the corresponding leak rate. Either way the circulators are designed to operate at the range of system pressures. The probability that forced cooling will also fail given that there is a helium leak is taken from the Modular High Temperature Gas Cooled Reactor PRA as 0.1. Therefore, the overall event frequency for small and large leaks coupled with a loss of blanket cooling is 3×10^{-4} and 10^{-5} per plant-year, respectively.

4.5.3. Accident Source Terms

The accident source terms were estimated by the ARIES/Starlite Team. The release of activation materials from the structure was estimated to be negligible by examining the release of sodium nuclides, ^{22}Na and ^{24}Na which are the dominant radio-nuclide as

the temperature of the first-wall and blanket structure is increased due to the afterheat during the accident. Therefore, the major source of radioactivity during the accident is the tritium.

There are two major sources of tritium. First tritium exists to a small degree in the coolant and in the purge gas as an impurity in the form of HT will be present. The amount is estimated as 1 to 5 g. This evaluation assumed 3 g available for release in seconds due to the blow-down.

Tritium is also produced in the beryllium multiplier at about one percent of the rate in the lithium breeder. Below 900 K the tritium will be retained in the beryllium and be available for release during the event. If the event is assumed prior to an annual heat-up to intentionally drive off and recover the tritium, there is about 400 g that would be released following the loss of blanket cooling in a time frame of 4 to 10 hours. This analysis assumed all 400 g released in 8 hours.

The above HT releases from the above sources were assumed to be released directly from the plant without retention or deposition in buildings. Ground level release was assumed. A building wake factor was used corresponding to building dimensions of 40 to 50 m high by 50 m wide and 100 m long.

An area of large uncertainty is the conversion of tritium, HT, in air to tritiated water, HTO. Based on experiments at Chalk River [7]] the conversion in air is not rapid (only about 1.5%/h), rather HTO forms in soils from deposited HT followed by reemission into the air. Thus, even after passage of the plume the low rates of HTO to HT persist. This is important since the dose effectivity of inhaled HTO is four orders of magnitude greater than for HT.

The meteorological data utilized was from a site used by EPRI on the advanced light water reactor program. The site encompassed 80%-85% of all US sites, that is, only 15%-20% of US sites would have more adverse weather conditions leading to higher off-site doses. The site data used one year worth of actual recorded data. A realistic treatment of population movement was employed and a realistic treatment of radionuclide settling from the building to the site boundary was considered.

The “best estimate” releases from both the small and large leak loss of cooling events were then statistically combined with the weather distribution using a Monte Carlo code. Figure 4.5-2 provides pictorially how the resulting dose distributions were combined and displayed as complementary cumulative distribution functions for comparison with the EPA PAG requirement at the plant boundary.

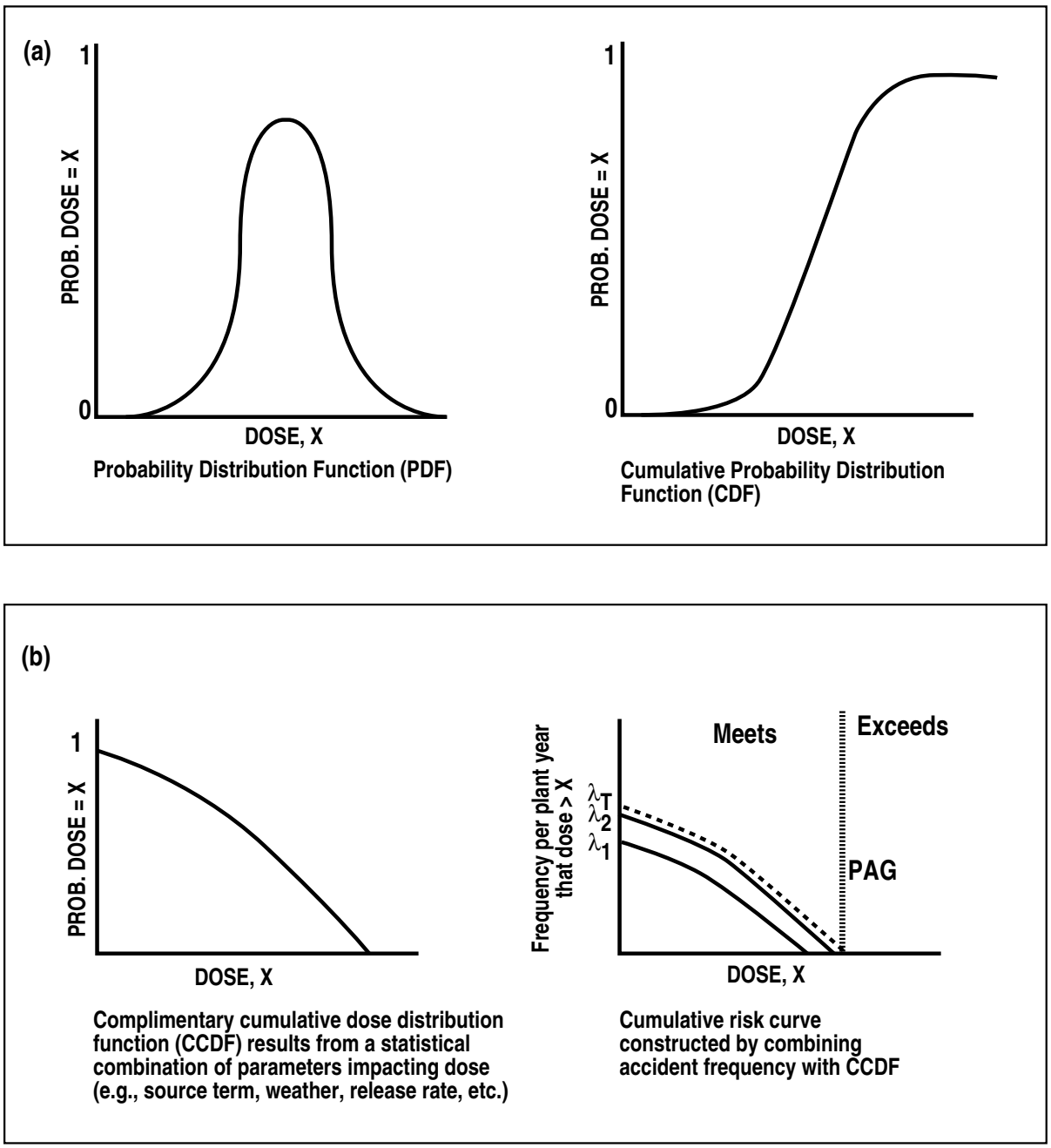

Figure 4.5-2. Description of probability distribution functions as first step in cumulative risk curve.

Figure 4.5-3 provides the preliminary ARIES-IV hazard assessment results. Two extremes are presented: one in which no tritiated water is formed and the other in which all tritium is converted to the more dominant dose contributor HTO. As previously stated the expected result is much closer to the "all HT" curves so that for this class of events the PAGs are expected to be met with significant margin. However, the "all HTO" results point out the importance of better understanding the conversion of HT to HTO. Note that in this abbreviated analysis only the uncertainty on the weather factored into the dose calculations. For example, the uncertainty in the source term was not included. Therefore, there is little sensitivity to the frequency of the two leaks, in fact, there is little importance to whether the leak is small or large. Again more detailed realistic modeling of the holdup in the buildings would likely decrease the small leak doses relative to those of the large leak.

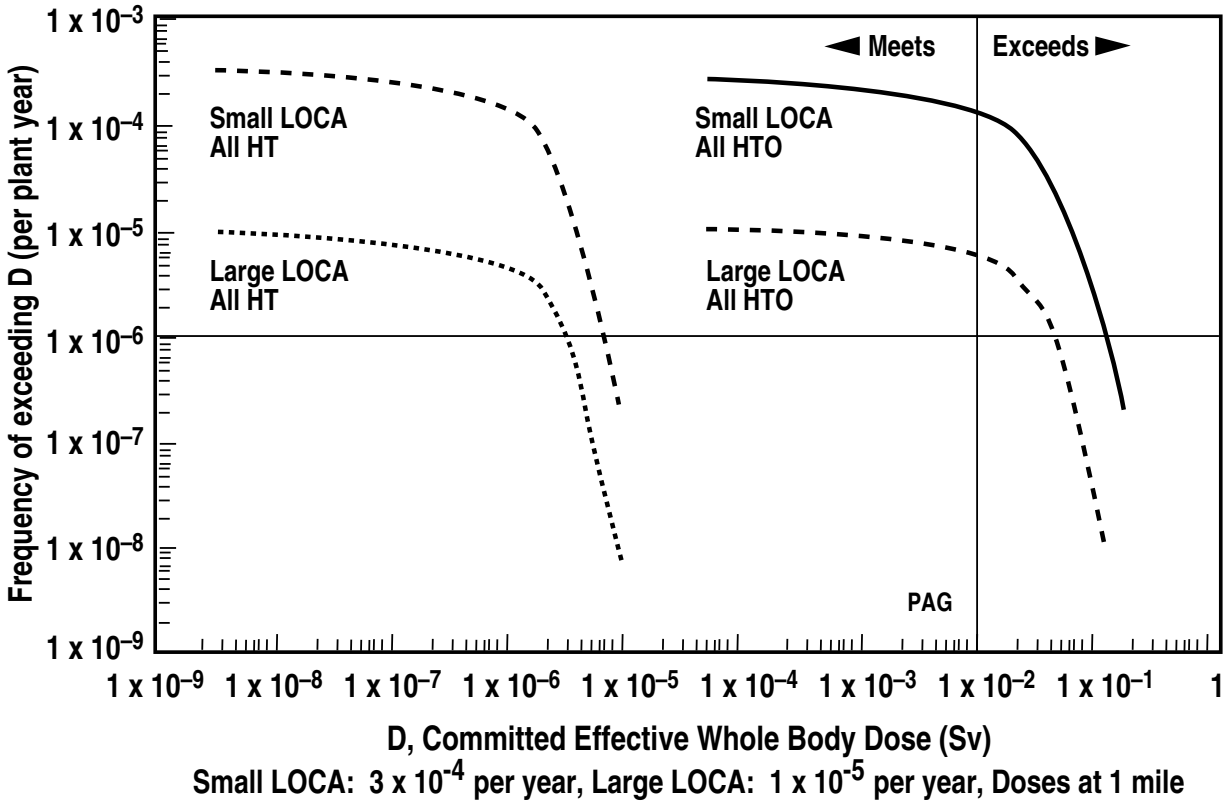

Figure 4.5-3. Preliminary ARIES-IV hazard assessment results compared to PAG (Break in the ARIES-IV helium-coolant line).

If, after a more comprehensive assessment with more events and more mechanistic consequence modeling is performed, greater margin is desired, three areas of the design could be adjusted as discussed below:

1. The accident source term could be reduced by more detailed quantification, by heating the beryllium blanket more frequently, or by reducing the plant power level;

2. The likelihood of the accident could be reduced by minimizing the number of coolant lines and/or their lengths, by providing welded versus bolted penetration closures, or by increasing the reliability of the forced helium cooling of the blanket;

3. The accident releases could be reduced by providing filtered release from the building, by providing elevated release (*i.e.,* a stack), by providing a retentive containment building, or by providing a larger plant boundary.

This preliminary hazard assessment selected primary coolant leaks and a loss of forced cooling as a representative event class to demonstrate the method for the ARIES-IV design. The study was not meant to definitely assess compliance with the off-site dose requirement, but rather to get "a feel for the lay of the land." All indications are that a more comprehensive assessment would show that compliance is achievable without major design changes or significant technology development primarily because the source term is limited, there are no internal sources of chemical energy, and the potential to instantaneously transport the source term from the plant is very low.

4.6. SUMMARY AND CONCLUSIONS

The major conclusions of the Safety and Licensing tasks are: (1) Review of NRC documentation yielded important insights in the context of fusion power systems; (2) A Strawman Pathway was proposed for the development of fusion regulatory policy and requirements; (3) Top-level safety and licensing requirements were established for fusion power systems; and (4) A methodology was established for demonstrating that sheltering and evacuation is not required for fusion power systems.

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