6.12 Safety and Environment

One of the paramount goals of the Prometheus project was to develop design criteria consistent with attaining a high degree of safety and compliance with all current and near term regulations. The approach taken to realize these goals was:

- Safety analysis focused on those aspects of Prometheus which are unique to an IFE reactor (i.e., differ from both a fission reactor and a magnetic fusion reactor).
- Approach and methodology from the ESECOM study[1] were used whenever feasible to allow for comparison of results to MFE.
- Designed Prometheus to meet (at a minimum) existing regulatory criteria.

Many of the safety and environmental analyses which were performed for specific Prometheus subsystems are presented in those sections of the report in which the particular subsystem is discussed. For example, Section 6.8.4 provides a description of the shielding analysis which was performed, while Section 6.7.10 provides a description of the calculation of tritium permeation through the heat exchangers that was completed. Section 6.8.2 discusses both an analysis which was done to determine the effects of dry spot formation on the wet wall and the analyses performed to determine an effective design for dissipating Pb^{203} decay heat following a loss of coolant accident.

Additionally, safety and environmental issues continuously influenced many features of the design, although no specific analysis may have been performed. For example, lead was selected as the first wall coolant because of the fire hazard associated with many of the other candidate metals (e.g., lithium, FLiBe). Also, the Prometheus-H reactor vessel was intentionally set as low as feasible within the reactor building in order to reduce seismic response, although a detailed seismic analysis was not performed. Because of its many beamlines, however, it was impracticable to place the Prometheus-L reactor vessel at a low elevation and additional structural support would have to be provided by a more robust reactor building.

Five areas in which specific safety and environmental assessment were made are:

- Occupational Radiation Exposures
- Off-Site Radiation Exposures
- Off-Site Impacts From Accidents
- Waste Disposal
- Non-Radiological Impacts

In this section, pertinent safety and environmental regulations from the Nuclear Regulatory Commission (NRC), the Environmental Protection Agency (EPA), the Department of Energy (DOE), and the Occupational Safety and Health Administration (OSHA) are reviewed for applicability to the Prometheus reactor. Due to the ground rule of this study which specified that the plant being analyzed is the tenth of a kind, the NRC's and EPA's regulations were used in most cases, instead of DOE's.

6.12.1 Occupational Radiation Exposures - The Nuclear Regulatory Commission's occupational radiation exposure limits are in the 21 May 1991 revision of 10CFR Part 20 and include the 1977 recommendations of the International Commission on Radiological Protection (ICRP), including those of ICRP Publication 26, and subsequent ICRP publications. The external occupational dose limits are:

Committed Effective Dose Equivalent	5 rem/yr
Lens of the Eye	15 rem/yr
Other Organ, Tissue, or Extremity	50 rem/yr

The 10CFR Part 20, Appendix B, Table 1 places limits on the airborne concentration of radionuclides within the work place. For Prometheus, tritium is anticipated to be the radionuclide of greatest concern during normal operations, and its derived air concentration from Appendix B, Table 1 is 2×10^{-5} μCi/ml. There is no difference between HTO and HT values because 10CFR Part 20 assumes that HT and T_2 will oxidize in the air and/or in the body to HTO.

In discussions with the NRC, it was recommended that, due to the time frame of this study, it would be appropriate to incorporate the 1990 recommendations of the ICRP (Publication 60) for normal operation.[2] Although Publication 60 specifies a maximum occupational dose of 5 rem/yr (consistent with current regulations), it also includes a basic dose limit of 10 rem per 5 years.

The Department of Energy's occupational radiation protection standards are contained in DOE Order 5480.11. As with the NRC's standards, these are based on the 1977 recommendations of the ICRP. Hence, the same annual occupational external dose limits as given above apply.

As with the NRC standard, the DOE requires that occupational doses be maintained as low as is reasonably achievable (ALARA). However, during the design of a facility, the DOE specifies that the use of design objectives which are 20% of the applicable standards will result in designs which are ALARA, without having to perform a "cost-benefit" analysis.

Additionally, the DOE regulations differ from the NRC's in the derived air concentrations. For tritium, the DOE provides a separate guide for airborne elemental tritium of 5×10^{-1} μCi/ml, in addition to an airborne HTO limit of 2×10^{-5} μCi/ml.

Prometheus has been designed to meet the occupational radiation exposure limits of both agencies and maintain exposures ALARA by not exceeding 20% of the applicable limit, as per DOE guidance.

6.12.1.1 Radiation Zone Definitions - For shielding design and access control purposes, the Prometheus plant has been divided into radiation access zones. Given below is a typical radiation zoning scheme that meets the requirements of the revised 10CFR Part 20. The association of areas to radiation zones is based on the occupancy requirements for each area (e.g., continuous occupancy is required in the Main Control Room, while some areas of the Auxiliary and Reactor Service Building will require daily access, and other areas of the building will only require access for maintenance). The necessity to keep exposures ALARA also factors into the assigning of radiation zones to the various areas (e.g., an area should be reduced to a Zone II from a Zone III if it can be done with a minimum expenditure, even though its occupancy requirements do not require that it be a Zone II).

Rather than perform a cost benefit of the shielding analysis to demonstrate the design is ALARA, the design external exposure limits for Zones II, III, and IV have been reduced to 20% of the applicable 10CFR Part 20 limit. The basis for this approach is the DOE Order 5480.11, Section 9.j.(1)(b), which states:

> External Radiation Exposure. The design objectives for personnel exposure from external sources of radiation in continuously occupied controlled areas are ALARA and not exceeding 0.5 mrem (5 microsieverts) per hour on average. The design objectives for exposure rates for potential exposure to a radiation worker where occupancy is generally not continuous are ALARA and not exceeding 20 percent of the applicable standard ...

These ALARA design objective dose rates for Zones II, III, and IV are given in parenthesis after the 10CFR Part 20 exposure limits.

Zone I: <0.05 mrem/hr - This zone has no restriction on occupancy. Such a zone would represent areas in the plant where radiation due to occupancy on a 40-hours per week, 50-weeks per year basis will not exceed the total effective dose equivalent to individual members of the public limit from 10CFR20.1301 of 100 mrem per year. Most non-employees and visitors to the site will receive considerably less than 100 mrem/yr because of the relatively short time interval during which they are on-site. Examples of Zone I areas include: Site Boundary, Administration Building, Turbine Building, etc.

Zone II: <2.5 mrem/hr (<0.5 mrem/hr ALARA Design Objective) - This zone is restricted to access by radiation workers only. Such a zone would represent areas in the plant continuously occupied by plant radiation workers and authorized visitors on a 40-hours per week, 50-weeks per year basis without exceeding the total effective dose equivalent occupational dose limit from 10CFR20.1101 of 5 rem per year. Examples of Zone II areas include: Main Control Room, Auxiliary and Reactor Service Building corridors, etc.

Zone III: <5 mrem/hr (<1 mrem/hr ALARA Design Objective) - This zone is restricted to access by radiation workers only. Such a zone would represent areas in the plant which could be occupied by plant radiation workers and authorized visitors on a less than 40 hours per week and/or less than 50 weeks per year basis without exceeding the total effective dose equivalent occupational dose limit from 10CFR20.1101 of 5 rem per year. The length of stay in this (and higher zoned areas) would be determined by the actual radiation level in the area, the past radiation history of the person entering, and the nature of the radiation. Examples of Zone III areas include: Auxiliary and Reactor Service Building, etc.

Zone IV: >5 mrem/hr (>1 mrem/hr ALARA Design Objective) - This zone is a restricted, radiation area as defined by 10CFR20.1003. Such a zone would need to be posted with a conspicuous sign(s) bearing the radiation symbol and the words "CAUTION, RADIATION AREA", in accordance with 10CFR20.1902(a). Examples of Zone IV areas include: Containment Building during shutdown, Radwaste System Tank cubicles, Auxiliary and Reactor Service Building valve galleries.

Zone V: >100 mrem/hr - This zone is a restricted, high radiation area as defined by 10CFR20.1003. Such a zone would need to be posted with a conspicuous sign(s) bearing the radiation symbol and the words "CAUTION, HIGH RADIATION AREA", in accordance with 10CFR20.1902(b). Access to areas within this zone would be controlled in accordance with the requirements of 10CFR20.1601. Examples of Zone V areas include: Containment Building during operations, Reactor Vessel area during shutdown.

Zone VI: >500 mrem/hr - This zone is a restricted, very high radiation area as defined by 10CFR20.1003. Such a zone would need to be posted with a conspicuous sign(s) bearing the radiation symbol and the words "GRAVE DANGER, VERY HIGH RADIATION AREA", in accordance with 10CFR20.1902(c). Access to areas within this zone would be controlled in accordance with the requirements of 10CFR20.1602. Examples of Zone VI areas include: Reactor Vessel area during operations.

Airborne Radioactivity - In addition to the above radiation zoning scheme, if an area is determined to contain, or potentially contain, airborne radioactivity in excess of the 10CFR Part 20, Appendix B derived air concentrations (DAC) (i.e., 2×10^{-5} μCi/ml for tritium), the suffix "A" will be added to its radiation zone designation (e.g., if a Zone III area contains airborne tritium in excess of the DAC, it would be designated as a Zone III-A area). Also, each airborne radioactivity area would need to be posted with a conspicuous sign(s) bearing the radiation symbol and the words "CAUTION, AIRBORNE RADIOACTIVITY AREA", in accordance with 10CFR20.1902(d).

6.12.1.2 In-Plant Radiation Monitoring System

The Prometheus In-Plant Radiation Monitoring System has been designed to inform operations personnel, both locally and in the Main Control Room, of radiation and radioactivity levels throughout the plant and to provide warning when abnormal levels occur in specific areas due to possible equipment malfunction. Some channels of the In-Plant Radiation Monitoring System are designed to Safety Class 1E requirements and can withstand accident environmental conditions. The safety class monitors are required to mitigate the effects of an accident by performing some action (e.g., isolate the Containment, close an exhaust valve). The In-Plant Radiation Monitoring System will consist of the following five subsystems:

- Area Radiation Monitoring System - Will measure the ambient radiation levels at various locations throughout the plant.
- Airborne Radiation Monitoring System - Will measure the airborne concentration of radionuclides at various locations throughout the plant.
- Tritium Airborne Radiation Monitoring System - Will measure the airborne concentration of tritium at various locations throughout the plant.
- Process Monitoring System - Will measure the concentration of radioactive fluids in the various process systems of the plant.
- Process Tritium Monitoring System - Will measure the concentration of tritium in the various process systems of the plant.

The In-Plant Radiation Monitoring System is divided into a safety class portion and a nonsafety-related portion, with all safety class equipment designed in accordance with IEEE 279-1971, IEEE 308-1980, IEEE 323-1983, IEEE 336-1985, IEEE 344-1987, and IEEE 384-1981. Safety class equipment will also be qualified in accordance with NUREG-0588 and Regulatory Guide 1.89 requirements.

Area Radiation Monitors - Safety class Area Radiation Monitors will include:

- Main Control Complex Area Monitor - To detect ambient levels of radiation in the control complex environment, particularly the Main Control Room.
- Containment Isolation Monitor - To detect increased levels of radiation in the exhaust air and to automatically isolate the ventilation system upon signal actuation.
- Containment Post-Accident Monitors - To detect levels of radiation in the containment during the post-accident period.

Non-safety Area Radiation Monitors will include monitors located in strategic areas of the plant (including the Laser Building and/or the Accelerator Tunnel) to indicate to operations personnel the radiation levels within the plant and to signal, alert, or alarm an unacceptably high level of radiation. On the order of 40 to 50 non-safety Area Radiation Monitors are expected to be installed in Prometheus.

Airborne Radiation Monitors - Safety Class Airborne Radiation Monitors will include:

- Main Control Complex Outside Air Intake Monitors - To detect increased radioactivity in the supply air to the control complex environment.

Non-safety Airborne Radiation Monitors will include monitors located within the plant to detect the presence of excess airborne concentrations of radioactivity either in cubicles, open areas, or ventilation ducts. On the order of 30 to 40 non-safety Airborne Radiation Monitors are expected to be installed in Prometheus.

The Airborne Radiation Monitors will measure particulates and noble gases. The sample is drawn into the monitor and then passes through a moving paper filter which collects particulates. A beta sensitive scintillation detector, aimed at the filter, monitors for particulate radiation. After passing through the filter, the sample is dried and then monitored for radioactive gas content.

Tritium Airborne Radiation Monitors - Safety class Tritium Airborne Radiation Monitors will include:

- Main Control Complex Outside Air Intake Monitors - to detect increased tritium levels in the supply air to the control complex environment.
- Target Factory - To detect increased tritium levels in the exhaust air and to automatically isolate the ventilation system upon signal actuation.

Non-safety Tritium Airborne Radiation Monitors will be collocated with the non-safety Airborne Radiation Monitors in those areas in which airborne tritium contamination could occur. Prometheus is expected to have on the order of 30 to 40 non-safety Airborne Radiation Monitors installed. The Tritium Airborne Radiation Monitors detect tritium airborne concentration by absorption and isotopic exchange processes.

Process Monitoring System - Both the Process Monitoring System and the Process Tritium Monitoring System are expected to contain radioactivity, as well as normally non-radioactive systems to detect developing equipment failure conditions. Twenty to twenty-five Process Monitors and Process Tritium Monitors are expected to be installed in the Prometheus design.

Process Tritium Monitors are similar to the Effluent Tritium Monitors, which are discussed in Section 6.12.2.3, below.

6.12.2 Off-Site Radiation Exposures - As stated above, the NRC's 21 May 91 revision of 10CFR Part 20 follows the lead of the 1977 recommendations of the ICRP, including those of ICRP Publication 26. For individual members of the general public, 10CFR20.1301 specifies an annual dose limit from normal plant operations of 0.1 rem total effective dose equivalent (TEDE).

In addition to the exposure limits contained in 10CFR Part 20 for nuclear power reactors, the NRC has numerical guides for design objectives and limiting conditions of operation contained in 10CFR Part 50, Appendix I. These limits were not revised to reflect the ICRP 1977 recommendations and remain based on the critical organ concept. The general public individual dose limits from effluents are:

Total Body - Airborne Effluent	5 mrem/yr
Any Organ - Airborne Effluent	15 mrem/yr
Total Body - Liquid Effluent	3 mrem/yr
Any Organ - Liquid Effluent	10 mrem/yr

In order to keep off-site exposures ALARA, 10CFR Part 50, Appendix I requires that all items of reasonably demonstrated technology that can effect reductions in dose to the population within 50 miles of the reactor be included in the design. The values of $1000 per total body man-rem and $1000 per man-thyroid-rem (based on 1977 dollars) should be used in determining whether an item is reasonably achievable.

In 10CFR Part 20, Appendix B, Table 2 places limits on the concentration of radionuclides in liquid and airborne effluents at the boundary of the unrestricted area. For Prometheus, tritium is the radionuclide of greatest concern during normal operations. The derived air concentrations for tritium are 1 x 10^{-7} μCi/ml for airborne effluents and 1 x 10^{-3} μCi/ml for effluents to water. As for occupational exposure, there is no distinction between HTO and HT Table 2 values because 10CFR Part 20 assumes that HT and T_2 will oxidize in the air and/or in the body to HTO.

The DOE's radiation protection standards and requirements for the public and the environment are contained in DOE Order 5400.5 and, since they are also based on the 1977 recommendations of the ICRP, the annual dose limit to members of the general public is the same as the NRC's 100 mrem.

As with occupational exposure, the DOE differs from the NRC in the tritium concentration guides. The DOE provides a separate guide for airborne elemental tritium of 2 x 10^{-2} μCi/ml, in addition to an airborne HTO limit of 1 x 10^{-7} μCi/ml. Also, because the NRC reduces its concentration limits by a factor of two so that they will be applicable to age groups other than adults, the DOE liquid tritium concentration guide of 2 x 10^{-3} μCi/ml is twice the NRC's value.

In 40CFR Part 61, "National Emission Standards for Hazardous Air Pollutants" (NESHAP), the Environmental Protection Agency (EPA) presents its regulations, under the Clean Air Act, for emissions of radionuclides. In 40CFR Part 61 sets standards for different categories of facilities, including DOE facilities, NRC-licensed facilities, and uranium fuel cycle facilities. However, when a dose limit is used as the standard, it is usually set at 10 mrem/yr (40CFR61.92 & 61.102) effective dose equivalent. Although the EPA applies NESHAP regulations to DOE facilities, it has suspended them for facilities (such as Prometheus) that are regulated by the NRC.

The EPA's regulations for radionuclides released into surface water are contained in 40CFR141.16, "National Primary Drinking Water Regulations, Mixed Containment Levels for Beta Particles and Photon Radioactivity from Man-Made Radionuclides in Community Water Systems." The dose limit in 40CFR141.16 for the maximum exposed individual is 4 mrem/yr.

Prometheus has been designed to meet the NRC's 10CFR Part 50, Appendix I off-site dose guides for design objectives and limiting conditions of operation, and limit the tritium concentration to the values given in 10CFR Part 20, Appendix B, Table 2. It will also, then, meet the applicable DOE and EPA requirements.

6.12.2.1 Normal Operation Release Estimate—HT vs HTO - The two primary forms of tritium released to the environment from Prometheus are tritiated water vapor (HTO) and tritium-hydrogen gas (HT), with a small amount of tritium gas (T_2). Since

HTO is approximately 25,000 times more radiotoxic than HT, it is important to know the fraction of the initial release which is HTO, as well as the extent of oxidation of HT to HTO in the open environment and the body, in order to realistically estimate the off-site exposure. However, little data are available regarding the rate of conversion of tritium from its gas form to tritiated water vapor. Two sources of such data are the experimental releases of HT, conducted at the Chalk River Meteorological Field on 26 August 1986 and 10 June 1987[3,4,5] and at the CEA Centre of Bruyere-le-Chatel on 15 October 1986.[6,7] These experiments confirmed that most of the oxidation occurs in the top layer of soil as opposed to in the atmosphere, and that the rate of conversion is slow. Russell and Ogram[5] present a maximum downwind HTO/HT concentration ratio of 3.4×10^{-4}, when pure HT is released.

Nonetheless, discussions with the NRC[2] indicate that because of the difficulty in determining when, where, and to what extent the oxidation of HT to HTO occurs, the NRC would be prone to assume that all tritium releases from the plant, both normal operation and accident conditions, are in the oxide form. As discussed above, this is consistent with revised 10CFR Part 20, Appendix B, which instructs that the HTO tritium annual limits on intake (ALI) and derived air concentrations (DAC) be used for HT and T_2, since "HT and T_2 oxidize in air and in the body to HTO."

With the assumption that all tritium is released in the HTO form, the maximum allowable annual tritium release from Prometheus can be determined from:

$$5\left[\frac{\text{mrem}}{\text{yr}}\right]=\frac{50\left[\frac{\text{mrem}}{\text{yr}}\right]}{1\times10^{-7}\left[\frac{\mu\text{Ci}}{\text{ml}}\right]}\bullet Q\left[\frac{\text{Ci}}{\text{yr}}\right]\bullet 1\times10^{-5}\left[\frac{\text{sec}}{\text{m}^3}\right]\bullet\frac{10^{-6}\bullet10^{6}}{3.15\times10^{7}}\left[\frac{\text{yr-m}^3\text{-}\mu\text{Ci}}{\text{sec-ml-Ci}}\right]$$

Where:

5 [mrem/yr]	=	The 10CFR Part 50, Appendix I whole body limit,
50 [mrem/yr]	=	The 10CFR Part 20, limit used to determine DACs,
1×10^{-7} [μCi/ml]	=	The 10CFR Part 20, Appendix B, Table 2, HTO DAC,
Q [Ci/yr]	=	Annual tritium release, and
1×10^{-5} [sec/m^3]	=	Annual average atmospheric dispersion.

All other parameters are units conversion factors.

Solving the above equation gives: $Q = 3.15 \times 10^4$ (Ci/y) = 86.2 (Ci/d). This is well above the daily tritium release design value of 10 [Ci/day] that was used in the Prometheus design (see Section 6.7). With a tritium release of 10 [Ci/day], the annual off-site dose would be less than 1 mrem. Taking credit for some tritium being in its elemental form will result in a greater maximum allowable annual release, while still maintaining off-site exposures within the 10CFR Part 50, Appendix I limits.

6.12.2.2 Effluent Monitoring System - The Prometheus Effluent Monitoring System measures the radioactivity concentration in the effluent streams from the plant for the determination of off-site releases. This system also actuates alarms and actions which prevent the continued effluent release if either the instantaneous or average release limits are exceeded. The Effluent Monitoring System monitors for particulate, noble gas, and tritium radioactivity at the point of release to the environs. The Effluent Monitoring System will consist of the following subsystems:

- Effluent Monitoring System - Will measure the concentration of radioactive fluids at the various release points of the plant.
- Effluent Tritium Monitoring System - Will measure the concentration of tritium at the various release points of the plant.

The following effluent points are measured for radioactivity:

- Plant Stack
- Containment Ventilation Exhaust
- Auxiliary Building Exhaust
- Target Factory Exhaust
- Laser Building Exhaust
- Accelerator Tunnel Exhaust
- Gaseous Radwaste Effluent
- Liquid Radwaste Effluent

The Effluent Tritium Monitors are composed of individual channels, consisting of a sampling ionization chamber, a check source, and an analyzer. The ionization chamber is ion differentiating so that non-tritium radionuclide emissions of a higher energy producing more energetic primary ions are spectrally differentiated from the weak tritium beta (i.e., 18.6 KeV). A Bremsstrahlung On-Line Monitoring System will be used for tritium monitoring of liquid effluents. Ionization chambers would be adequate for gaseous effluents. This system is comprised of a dual matched GELI detector system with two bremsstrahlung windows, one of gold foil and the other of aluminum foil. By virtue of a subtract mode, the differential signal between the gold-foiled GELI and aluminum-foiled GELI yields the actual tritium bremsstrahlung signal, since all other signals would be registered on both detectors with the same magnitude.

6.12.3 Off-Site Impacts From Accidents - For the purposes of defining the exclusion area of a power reactor, 10CFR Part 100 specifies that for the two hours immediately following onset of the postulated accident, an individual located at the exclusion area boundary would not receive a whole body dose in excess of 25 rem or a thyroid dose in excess of 300 rem. Likewise, for determining the distance to the boundary of the low population zone, 10CFR Part 100 specifies that for the duration of the postulated accident, an individual located at the low population zone boundary would not exceed the same limits. The postulated radionuclide release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from consideration of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible.

In addition to the guidance provided in 10CFR Part 100, the NRC has developed Standard Review Plans (NUREG-0800)[8] that describe how the NRC reviews each section of a Safety Analysis Report, including the design basis accident analyses of each system and subsystem. Depending on the expected frequency of occurrence of an accident, NUREG-0800 requires that the calculated off-site exposures be either "within," "well within (i.e., 25%)," or "a small fraction of (i.e., 10%)" 10CFR Part 100 limits.

In DOE Order 6430.1A, "Design Criteria," Section 0200-1.3 "Radiological Siting Guidelines," the DOE provides dose values to be used in the evaluation of design basis accidents. In addition to the whole body and thyroid dose limit (which are the same as the NRC's 10CFR Part 100 limits), Order 6430.1A provides limits for the bone surface (300 rem), the lung (75 rem), and other organs (150 rem). Finally, this Order specifies an effective dose equivalent limit (25 rem) in keeping with the ICRP's 1977 recommendations.

Prometheus was designed to meet the DOE's effective dose equivalent limit of 25 rem for the worst case design basis accidents and will be "well within" or a "small fraction of" this limit for lesser, more frequent design basis accidents. It is not necessary to impose the thyroid dose limit on Prometheus, since radio-iodine will not be a contributor to exposures from a fusion reactor.

6.12.3.1 **LSA Determination** - The ESECOM study,[1] sponsored by the DOE, is the most comprehensive examination of safety and environmental issues for magnetic fusion energy (MFE) reactors. Recognizing that the current state of development of fusion reactor technology is still in the conceptual design phase and that high levels of uncertainty exist about postulated accident sequences and probabilities, MFE studies could not follow the path taken by the fission reactor community, where extensive (and expensive) PRA accident analyses are conducted during the design phase of each new plant. Instead, the ESECOM study applied a simpler, semi-quantitative approach to safety analysis of MFE reactors. This same methodology was applied to the Prometheus reactors.

The ESECOM study defined four levels of safety assurance (LSA) to characterize the relative risks of different design concepts. Although the categories are coarse, such classification facilitates comparison of the degree of inherent safety (i.e., passive versus active safety systems) in different IFE designs and, indeed, comparison to MFE designs (e.g., ARIES) and fission reactors. LSA levels are defined as:

Level	Characteristics
1	*Safety is assured by passive mechanisms of release limitations no matter what the accident sequence. The radioactive inventories and materials in such a reactor preclude a fatal release regardless of the reactor's condition.*
2	*Safety is assured by passive mechanisms of release limitations as long as severe reconfiguration of large-scale geometry is avoided, and escalation to fatality-producing reconfigurations from less severe initiating events can plausibly be precluded by passive design features. In such a reactor, natural heat-transfer mechanisms suffice to keep temperatures below those needed–given its radioactivity inventories and material properties–to produce a fatal release unless large-scale geometry is badly distorted.*
3	*Safety is assured by passive mechanisms of release limitations as long as severe violations of small-scale geometry–such as a large break in a major coolant pipe–are avoided, and escalation to fatality-capable violations from less severe initiating events can plausibly be precluded by passive design features. In such a reactor, sufficiency of natural heat-transfer mechanisms to keep temperatures low enough to avoid a fatal release–given the radioactivity inventories and materials properties–can only be assured while the coolant boundary is substantially intact.*
4	*There are credible initiating events that can lead to fatalities without any severe violation of small-scale or large-scale geometry or that can only be prevented from escalating to fatality-capable boundary violations or reconfigurations by means of active safety systems.*

Thus, the LSA categories range from totally passive (LSA = 1) to conditions requiring active safety features (LSA = 4). The ESECOM study also developed the concept of threshold dose release fractions (TDRF) to evaluate the potential hazard of a particular radionuclide appearing in a particular material or form. The TDRF of a radionuclide is defined as that fraction which must be mobilized to produce an off-site dose equal to some dose limit. For TDRF greater than one, release of all the inventory would not exceed the dose limit.

The threshold dose release (i.e., the release which would result in an off-site individual receiving a dose equivalent to the dose criteria) was determined for Prometheus from the ESECOM data. These are presented on Table 6.12-1. The N^{16} ESECOM

Table 6.12-1 ESECOM[1] Threshold Dose Release (Curies)

	Case 1	Case 2
H^3	2.5×10^8	8.9×10^7
N^{16}	—	—
Al^{28}	1.5×10^{11}	—
Cr^{51}	1.8×10^8	1.9×10^6
Fe^{55}	6.4×10^8	1.1×10^7
W^{187}	2.4×10^7	5.4×10^6
Pb^{203}	3.1×10^7	3.3×10^6
Pb^{209}	2.1×10^9	5.5×10^6
Po^{210}	4.2×10^4	9.6×10^3

threshold release was not determined due to the extremely short half-life of N^{16}. The Prometheus inventories (found in Section 6.4 and 6.7 for tritium and Section 6.8 for the other radionuclides) were compared to the release values and found to be less for all radionuclides (i.e., TDRF > 1.0), with the exception of W^{187} and Pb^{203}.

For example, it has been determined that the Prometheus Po^{210} inventory in the lead coolant amounts to a maximum of 1.6×10^3 Ci. The ESECOM limit for this radionuclide, so as not to exceed an off-site dose (case 1) of 200 rem, is 4.2×10^4 Ci. Thus, the TDRF = $4.2 \times 10^4 / 1.6 \times 10^3 = 26$. Since this is much greater than 1.0, even if all the Po-210 were to become mobilized, the off-site dose would not exceed the regulatory limit. This results in an LSA of one (totally passively safe) for all accidents involving only Po^{210} mobilization.

With respect to the two Prometheus radionuclides with a TDRF of less than one, the following observations are made. The ESECOM W^{187} inventory also exceeded that which would allow classification as an LSA of one. However, ESECOM still classified the SiC reactor system as a nominal LSA of one because of the low inherent mobility of tungsten (an ESECOM mobility category of IV "somewhat volatile"), and the absence of a mechanism for mobilization of the tungsten. Likewise, the W^{187} component of the Prometheus design will be classified as an LSA of one.

A larger concern is the Pb^{203}, since its inventory exceeds the Threshold Dose Release value and since lead has been classified into a higher mobility category then tungsten (an ESECOM mobility category of III "somewhat to highly volatile"). Also, as discussed in Section 6.8.2, the Prometheus design has had to incorporate features to dissipate the decay heat generated by the Pb^{203} which could be disrupted by a reconfiguration of large-scale geometry. For these reasons, the tentative classification of Pb^{203} as an LSA of one is subject to change, but in any case the highest LSA classification for Pb^{203} would be that of two.

This LSA determination demonstrates the technique of radionuclide "inventory control" where passive safety is assured by limiting amounts of radionuclides below quantities which could exceed off-site dose limits if mobilized. The Prometheus design specifies low activation materials where possible (e.g., SiC structural material, He coolant) and limits amounts of tritium contained in any one place.

6.12.3.2 <u>NRC Compliance</u> - A second evaluation was performed to demonstrate compliance with the above quoted NRC and DOE regulations using the methodology specified by the NRC in various Regulatory Guides, Standard Review Plans, and elsewhere.

Because the radionuclides of interest for this study are not the common radionuclides of interest for a fission reactor, the fission literature does not provide readily available dose conversion factors. Therefore, the dose conversion factors for this evaluation were determined from DOE Order 5400.5 derived concentration guides (DCG) and dose limits. The dose conversion factors which were determined, as well as the DCGs, are provided on Table 6.12-2.

The radionuclide release which will result in a specified off-site dose can be determined from:

$$R\,[\mathrm{Ci}] = \frac{D[\mathrm{rem}]}{\frac{\chi}{Q}\left[\frac{\mathrm{sec}}{\mathrm{m}^3}\right] \bullet \mathrm{DCF}\left[\frac{\mathrm{rem\text{-}m}^3}{\mathrm{Ci\text{-}sec}}\right]}$$

Where:

R [Ci]	=	the NRC threshold dose release,
D [rem]	=	the 10CFR Part 100 dose limit,
χ/Q [sec/m^3]	=	the atmospheric dispersion, and
DCF [(rem-m^3) / (Ci-sec)]	=	the dose conversion factor.

The atmospheric dispersion was determined to be 1.78 x 10^{-4} [sec/m^3] via the plume meander methodology from Regulatory Guide 1.145 and the meteorological parameters from the ESECOM study (i.e., a site boundary distance of one kilometer, stability class F, and a wind speed of 1 meter/second). The NRC threshold dose release values that were determined are presented in Table 6.12-2. For example, for tritium:

$$4.43 \times 10^{6} = \frac{25}{1.78 \times 10^{-4} \bullet 3.17 \times 10^{-2}}$$

Note that the calculated NRC threshold dose releases are much lower than the ESECOM threshold dose releases. This is due to the different methodology utilized in their determinations and is not contradictory since the NRC allows credit to be taken for other factors such as hold up by the containment.

Table 6.12-2 NRC Compliance Parameters

	Derived Concentration Guides Inhalation (mCi/ml)	Derived Concentration Guides Immersion (mCi/ml)	DCF (rem-m^3) (Ci-sec)	NRC Threshold Dose Release Curies)
H^3 (HTO)	1.0 x 10^{-7}	Not Given	3.17 x 10^{-2}	4.43 x 10^{6}
N^{16}	Not Given	3.0 x 10^{-9}	1.06 x 10^{0}	—
Al^{28}	1.5 x 10^{11}	1.0 x 10^{-8}	3.17 x 10^{-1}	4.43 x 10^{5}
Cr^{51}	5.0 x 10^{-8}	Not Given	6.34 x 10^{-2}	2.22 x 10^{6}
Fe^{55}	5.0 x 10^{-9}	Not Given	6.34 x 10^{-1}	2.22 x 10^{5}
W^{187}	2.0 x 10^{-8}	Not Given	1.58 x 10^{-1}	8.89 x 10^{5}
Pb^{203}	2.0 x 10^{-8}	Not Given	1.58 x 10^{-1}	8.89 x 10^{5}
Pb^{209}	1.0 x 10^{-7}	Not Given	3.17 x 10^{-2}	4.43 x 10^{6}
Po^{210}	1.0 x 10^{-12}	Not Given	3.17 x 10^{3}	4.43 x 10^{1}

6.12.4 Waste Disposal - Waste disposal issues were also investigated, considering both radioactive and hazardous (non-radioactive) waste regulations as applied to Prometheus. The Resource Conservation Recovery Act (RCRA) is the primary Federal statute which governs the regulation of solid and hazardous waste. RCRA is broken into many subtitles. The subtitle particularly pertinent to Prometheus is C: "Hazardous Waste." Subtitle C provides "cradle-to-grave" regulation of hazardous waste and authorizes the EPA to regulate the generation, transportation, treatment, storage, and disposal of hazardous waste. Regulations governing these processes are contained in 40CFR Part 262 (generation), 40CFR Part 264 (treatment, storage, and disposal), 40CFR Part 268 (treatment, and land disposal), and 49CFR Part 170 (transportation).

Low level radioactive waste (LLW) is regulated by the NRC under 10CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste." Depending on the degree of radioactivity, 10CFR 61.55 classifies waste as being Class A, Class B or Class C, with Class A waste being the least radioactive. It is expected that Prometheus, like other proposed IFE reactor plants, along with MFE plants, should produce a much lower level of radioactive waste than the same size fission plants.

"Greater than Class C" waste is waste with radionuclide concentrations which are greater than those allowed by 10CFR Part 61 for disposal in a near surface facility, but which do not meet 10CFR Part 60 definition of high level waste. In a Statement of Consideration, the NRC has stated that all greater than Class C waste is a Federal responsibility and indicates that no permanent disposal facility for this waste will be available for at least 15 to 20 years. In the meantime, the NRC suggests that the Federal government provide limited access to an existing DOE storage facility.

The only Prometheus radionuclide of concern that has specific concentration limits specified in 10CFR61.55, Table 1 or 2 is C^{14}. The 10CFR61.55 specifies that if the C^{14} concentration in waste does not exceed 0.8 Ci/m^3 then the waste is Class A. If the C^{14} concentration is greater than 0.8 Ci/m^3, but less than 8 Ci/m^3, then the waste is Class C. The C^{14} concentration in Prometheus ranges from 0.13 Ci/m^3 after two years of full power operation in the first wall to 0.2 Ci/m^3 after 30 years of full power operation in the blanket. (See Table 6.8.5-2 and related text for explanatory data.) Therefore, the **first wall and blanket can be disposed of as Class A LLW.**

The regulations contained in 10CFR Part 61 were developed by the NRC to address existing LLW concerns and only explicitly address a very limited number of radionuclides. If a fusion reactor design using liquid lead as a coolant becomes a popular power reactor, then the NRC may revise their regulation to address radionuclides which are characteristic of this design (e.g., Pb-205). Reference 9 provides an estimate of what LLW concentration limits could be for radionuclides of interest to fusion reactors, including Pb-205. In determining these limits, Reference 9 utilized the same methodology that the NRC used to determine the 10CFR Part 61 limits. For Pb-205, Table II of Reference 9 gives a concentration limit of 1×10^6 Bq/cm^3 (27 Ci/m^3) but does not indicate whether this is a Class A or Class C limit. The maximum Prometheus Pb-205 concentration is 13.2 Ci/m^3 after two years of full power operation, or 198 Ci/m^3 after 30 years of full power operation.

Mixed wastes are wastes which contain both hazardous and radioactive materials. The 10CFR61.2 defines hazardous waste as "those wastes designated as hazardous by EPA regulations in 40CFR Part 261." The 10CFR61.56 states that any radioactive waste "containing hazardous, biological, pathogenic, or infectious material must be treated to reduce to the maximum extent practicable the potential hazard from the non-radiological materials." The 40CFR Part 261 considers lead to be an extraction procedure (EP) toxic substance, as per the Extraction Procedure Toxicity Test (40CFR Part 261, Appendix II) and has given lead the hazardous waste number D008 (40CFR261.24, Table I). For Prometheus, the lead coolant will have to be classified as a hazardous material and any radioactive lead which cannot be recycled will have to be disposed of as mixed waste (both radioactive and hazardous).

The Hazardous and Solid Waste Amendments Act of 1984 brought mixed radioactive/hazardous waste into the RCRA universe. Prometheus will need RCRA generator permits for mixed waste under 40CFR Part 262.

6.12.4.1 __Waste Volume Estimates__ - As described in Section 6.8.2, the first wall of Prometheus has been designed to last two years. With the replacement of the first wall, the old wall will have to be disposed of as LLW. The volume of SiC in the first wall is approximately 170 ft^3, assuming a packing fraction of 0.67 (to account for fitting the shaped wall into a disposal container). The amount of first wall to be disposed of is estimated to be 255 ft^3 over two years (or 128 ft^3/yr). For the purpose of this study, a LLW disposal cost of $100 per cubic foot has been assumed.

The amount of lead contained in the first wall coolant system is estimated at 540 ft^3. Because of the high cost of disposal of mixed waste (currently estimated to be $15,000 per cubic foot[10]), it is preferable to keep the original lead as the first wall coolant for the duration of operations (i.e., 30 years). After operations cease, the lead can either be disposed of as mixed waste or recycled into the replacement reactor. With the recycle option, however, the Pb^{205} will continue to increase in concentration due to its extremely long half-life (15 million years).

6.12.5 Non-Radiological Impacts - The Occupational Safety and Health Administration (OSHA) has promulgated a voluminous set of health and safety standards. OSHA's standards, which apply to employee activities in all industries (including the electric utility industry), are codified as 29CFR Part 1910. OSHA's standards for construction activities (including power plant construction) are codified as 29CFR Part 1926. In general, these standards have already been adopted by nationally recognized standards setting organizations and Federal standards in existence in 1971. In addition to complying with these standards, employers are required to maintain records of employee exposure to potentially hazardous materials and report to OSHA periodically work-related deaths, injuries, and illness.

Lead toxicity and exposure are of interest in the Prometheus design, since lead is used as a first wall coolant and protector. The regulations governing occupational exposure to airborne lead are defined by OSHA and are contained in 29CFR1910.1025, which defines an action level concentration of 30 $\mu g/m^3$, time weighted average (TWA), based on an 8-hour work day. The action level initiates several requirements, such as exposure monitoring, medical surveillance, and training and education. The permissible exposure limit for lead is defined as 50 $\mu g/m^3$ TWA. If an employee is exposed to lead above the permissible level, a respirator must be worn by the employee. In addition, when the lead level exceeds the permissible exposure limit, warning signs must be posted.

The National Institute for Occupational Safety and Health (NIOSH) has set an Immediately Dangerous to Life or Health (IDLH) level for lead of 700 $\mu g/m^3$. The Control Room personnel (and other critical personnel) will be protected such that the lead levels are below this IDLH value for all credible accidents.

Also of special concern in the Prometheus-L plant is the possibility of exposure of personnel to fluorine in the KrF laser system. The NIOSH has set the time weighted average exposure limit for fluorine at 0.1 ppm (0.2 $\mu g/m^3$), with an IDLH value of 25 ppm. The time weighted average exposure limit for fluorides is set at 5.5 $\mu g/m^3$, with an IDLH value of 500 $\mu g/m^3$.

The Clean Water Act of 1977 established a comprehensive program to regulate pollutant discharges into surface waters of the United States. This program is implemented primarily through the discharge permit program of CWA Section 402, called the National Pollutant Discharge Elimination System (NPDES). The Environmental Protection Agency (EPA) usually delegates to the states the enforcement of the NPDES. Since for this study a particular state has not been identified, a "representative" set of discharge limits is postulated based on 40CFR Part 423.

Detailed analysis of Prometheus for compliance with non-radiological regulations was not performed. However, no "design threatening" problems were found or expected to be found following a future, more comprehensive examination.

6.12.6 Safety and Environment Conclusions - The Prometheus study has identified applicable regulations which must be factored into the design of a commercially viable IFE power plant and demonstrated that they can be met. The ESECOM level of safety assurance methodology was applied to Prometheus in assessing off-site doses due to releases of key radionuclides present in the plant. With the exception of W^{185} and Pb^{203}, the Prometheus inventories allow the plant to be classified as totally passively safe (LSA = 1). The tungsten isotope was also identified in the ESECOM study, but not seen as a problem due to its immobility. The lead isotope, however, is unique to Prometheus with its lead coolant, and design features to remove lead afterheat in the event of loss of cooling have been incorporated to address this issue.

In addition to determining the LSA level of Prometheus, compliance with the NRC's accident dose criteria and the NRC's and EPA's normal operational dose criteria were also determined. Environmental, nonradiological EPA regulations such as the Clean Water Act and the Resource Conservation Recovery Act were examined with no "design threatening" problems identified or expected to be identified.

Finally, during the course of the Prometheus study, a number of safety and environmental issues were identified that must be addressed prior to the successful commercial operation of IFE reactors. These issues are identified and discussed in Section 5.3 Key Issues Description. Although some of these safety and environmental issues are a direct result of the Prometheus design (e.g., liquid lead), others are generic to Inertial Fusion Energy (e.g, local dry spots), while still others are generic to all fusion reactors (e.g., tritium). With few exceptions, these safety and environmental issues can be resolved via engineering analysis and design and do not constitute a major obstacle to the successful development of IFE.

References for 6.12

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8. U.S. Nuclear Regulatory Commission, "Standard Review Plan," NUREG-0800.

9. C. Ponti, "Recycling and Shallow Land Burial as Goals for Fusion Reactor Materials Development," Fusion Technology, Volume 13, January 1988.

10. D. W. Stever, "EPA Regulation of Mixed Waste," Sidley & Austin, Mixed Waste Regulation Conference, June 1991.

6.13 Economics

This section presents an economic assessment (cost analysis) for the Prometheus power plants. The projected costs are developed using the basic groundrules and assumptions discussed in Section 3, together with the parametric performance and cost scaling developed in support of the systems modeling. Efforts have been made to normalize the cost projections across all subsystems, but this is difficult because proposed technologies often correspond to significant extrapolations of present-day hardware. Where possible, costs were normalized using assumptions from recent MFE studies[1-2] to provide a common basis for comparison. Elsewhere, costs were based on the best judgment of technical experts. Economic scaling in the present systems model ICCOMO has evolved over many years. The models were originally developed as part of the STARFIRE reactor design study[3] and were adapted to IFE as part of the HIFSA project[4]. As a result, many of the cost models used here have achieved some level of acceptance within the fusion community. A detailed description of the final cost models is presented in Appendix C.

The program guidelines stated that costs are to be evaluated for a tenth-of-a-kind power plant. The enclosed economic analysis has attempted to conform to this guideline and the rationale for doing so is presented in Appendix C. However, cost scaling models in the systems code are represented in terms of the first commercial unit. This prevents a somewhat arbitrary learning curve cost adjustment from influencing the systems trade studies. Many results in this section are, therefore, presented in the form of relative comparisons in order to avoid confusion in relating them to the tenth unit costs discussed for the baseline design point. In cases where learning curve adjustments have been applied, they are so noted and the assumed learning factor is documented.

The effect of different learning curves on projected cost is shown in Figure 6.13-1. It compares unit cost reduction factors for 90, 80 and 70% learning curves in plants with a single unit per plant to those for plants with 100 units per plant. This figure highlights the significant effect that learning curve assumptions can have on the projected COE for the tenth plant. For example, the cost of the tenth plant reactor vacuum vessel (one per plant) is less than half of its first plant cost assuming an 80% learning curve. On the other hand, the cost of grazing incidence mirrors (60 per plant) will be less than 40% of the first plant cost for the same learning curve assumption. Therefore, experience and judgment were used in applying learning curves to the unit costs with values of 85-90% typically assumed for most systems.

A cost summary for the Prometheus-L and -H tenth-of-a-kind power plants is presented in Table 6.13-1. This table also shows the learning curves assumed for each subsystem. Note that learning curves were used only in the target plant, reactor plant, and driver systems and not to the "standard" balance-of-plant systems.

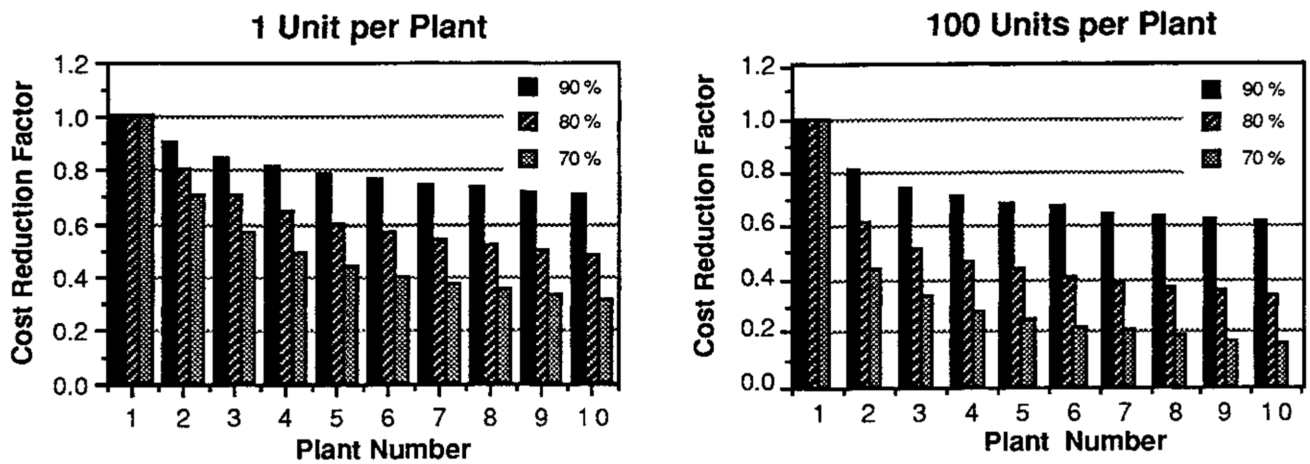

Figure 6.13-1. Unit Cost Reduction Factors for 70, 80 and 90% Learning

Comparing the two cost summaries of Table 6.13-1 shows that the projected COE for the laser-driven plant is ~15% higher than that for the heavy ion linac-driven plant. This is a surprising result for a 1000 MWe power plant since it is contrary to the past perceptions that heavy ion drivers were only attractive for larger units (> 1500 MWe) where the high capital cost of the driver could be distributed over more delivered power. The innovative single beam linac results in a heavy ion driver cost that is nearly identical to that for the laser (~$400M) even though the laser delivers only 4 MJ compared to 7.8 MJ for the heavy ion system. The COE difference is, thus, a direct consequence of ηG which is 21 for the heavy ion compared to only 8 for the laser system. As a result, the reactor and balance-of-plant systems must be larger for the laser driver and this is reflected in their costs which are ~$230M higher for the laser plant. In addition, as discussed in Section 6.3, the projected plant availability is 80.8% for the heavy ion system as compared to 79.4% for the laser which also lowers the heavy ion COE.

A key aspect of this cost difference involves the ~$130M reduction in reactor plant cost for the heavy ion design in spite of a fusion yield of 719 MJ as compared to 497 MJ for the laser system. The design studies indicate that cavity vapor pressure, not target yield, is the determining factor for the design of wetted wall cavities larger than 4 m radius. This permits the use of nearly identical cavity designs for both driver concepts. Cavity vapor pressure and size are directly related for a wetted wall design. This is due to the need to conduct surface heat through the first wall structure and into the coolant while maintaining a suitable surface temperature and cavity vapor pressure.

Table 6.13-1. Cost Summaries for Prometheus-L and -H Power Plants

COST SUMMARY FOR 972 MWe PROMETHEUS-L DESIGN POINT

1991$ Interest=16.5% Escalation=0.0% Fixed Charge Rate=9.7% Construct Time=6.0 yrs Avail=0.794 LSA=1

		Lrn Crv			Lrn Crv		CYD	NYD
LAND	**10.00**	None	**REACTOR PLANT**	**700.43**		**DIRECT COST**	**2171.67**	**2171.67**
			Equipment	391.82				
STRUCTURES	**376.66**	None	FW Blanket	157.03	85%	**CONSTR SERVICES**	**245.40**	**245.40**
Site Improvements	21.00		Bulk Shield	203.82	90%			
Reactor Building	106.06		Structure	22.99	85%	**ENG - HOME**	**112.93**	**112.93**
Driver Building	36.52		Vacuum System	4.54	85%			
Turbine Building	57.18		Target Injection	3.44	85%	**ENG - FIELD**	**112.93**	**112.93**
Heat Rejection	11.48		Heat Transfer	198.59	75%			
Auxiliary Power	15.54		Auxiliary Cooling	4.87	85%	**OWNER COST**	**396.44**	**396.44**
Target Manufacturing	46.92		Waste Processing	5.68	None			
Tritium & Waste Proc	46.92		Fuel Handling	48.15	85%	**PRCSS CONTGNCY**	**0.00**	**0.00**
Misc Structures	35.03		Maint Equipment	21.07	85%			
			Reactor I&C	26.60	90%	**PRJCT CONTGNCY**	**445.27**	**445.27**
			Other	3.65	None			
TURBINE PLANT	**327.13**	None	Spare Parts	0.00		**INTEREST**	**575.66**	**1107.41**
			Contingency	0.00				
ELECTRIC PLANT	**165.46**	None				**ESCALATION**	**0.00**	**848.86**
			DRIVER	**398.28**				
MISC PLANT	**57.15**	None	Front End	10.25	90%	**TOTAL COST**	**4060.29**	**5440.90**
			Discharge Lasers	44.91	90%			
TARGET PLANT	**134.94**	90%	RAC System	4.65	90%			
			SBS System	1.27	90%	**ANNUAL CAPITAL**	**392.22**	**891.22**
SPCL MATERIALS	**1.62**	None	Gas Flow System	33.91	80%			
			Optics	106.18	90%	**OPERATING COST**	**94.96**	**127.25**
			Pulsed Power	142.14	90%	Plant O&M	49.71	66.61
			Vacuum System	0.00	90%	Sched Replacemt	24.29	32.55
			I&C	20.00	90%	Target Mfg O&M	12.32	16.51
			Maint Equipment	16.66	85%	Fuel	4.08	5.47
			Misc Equipment	18.31	90%	Decommission	4.56	6.11
			Spare Parts	0.00				
			Contingency	0.00		**COST OF ELECT**	**72.0**	**150.5**

COST SUMMARY FOR 999 MWe PROMETHEUS-H DESIGN POINT

1991$ Interest=16.5% Escalation=0.0% Fixed Charge Rate=9.7% Construct Time=6.0 yrs Avail=0.808 LSA=1

		Lrn Crv			Lrn Crv		CYD	NYD
LAND	**10.00**	None	**REACTOR PLANT**	**568.34**		**DIRECT COST**	**1940.64**	**1940.64**
			Equipment	271.64				
STRUCTURES	**322.41**	None	FW Blanket	129.62	85%	**CONSTR SERVICES**	**219.29**	**219.29**
Site Improvements	21.00		Bulk Shield	129.58	85%			
Reactor Building	60.88		Structure	6.19	85%	**ENG - HOME**	**100.91**	**100.91**
Driver Building	38.27		Vacuum System	3.51	85%			
Turbine Building	52.79		Target Injection	2.74	85%	**ENG - FIELD**	**100.91**	**100.91**
Heat Rejection	10.60		Heat Transfer	182.70	75%			
Auxiliary Power	14.81		Auxiliary Cooling	3.52	85%	**OWNER COST**	**354.26**	**354.26**
Target Manufacturing	44.47		Waste Processing	5.41	None			
Tritium & Waste Proc	44.47		Fuel Handling	55.38	85%	**PRCSS CONTGNCY**	**0.00**	**0.00**
Misc Structures	35.11		Maint Equipment	19.44	85%			
			Reactor I&C	26.60	90%	**PRJCT CONTGNCY**	**397.90**	**397.90**
			Other	3.65	None			
TURBINE PLANT	**282.98**	None	Spare Parts	0.00		**INTEREST**	**514.42**	**989.60**
			Contingency	0.00				
ELECTRIC PLANT	**151.59**	None				**ESCALATION**	**0.00**	**758.55**
			DRIVER	**403.11**				
MISC PLANT	**57.27**	None	Ion Source	7.05	90%	**TOTAL COST**	**3628.34**	**4862.07**
			Accelerator Structure	70.84	90%			
TARGET PLANT	**143.62**	90%	Focusing Magnets	43.25	90%			
			Cryogenic System	8.67	90%	**ANNUAL CAPITAL**	**350.50**	**796.41**
SPCL MATERIALS	**1.32**	None	Storage Rings	13.05	80%			
			Final Transport	53.72	90%	**OPERATING COST**	**92.52**	**123.98**
			Pulsed Power	150.98	90%	Plant O&M	50.39	67.53
			Vacuum System	9.66	90%	Sched Replacemt	20.17	27.03
			I&C	19.00	90%	Target Mfg O&M	11.80	15.81
			Maint Equipment	19.29	85%	Fuel	5.60	7.50
			Misc Equipment	7.60	90%	Decommission	4.56	6.11
			Spare Parts	0.00				
			Contingency	0.00		**COST OF ELECT**	**62.6**	**130.0**

The cavity size is actually smaller for the heavy ion system, 4.5 compared to 5 m, due to several factors. The primary factor is the proposed self-pinched, heavy-ion transport channel (see Section 4.3.2). This decouples beam focusing concerns from the cavity pressure environment, permitting relative freedom in choosing a heavy ion cavity operating pressure (100 mtorr compared to < 5 mtorr) for the laser where laser induced gas breakdown is a concern. Fusion power is also reduced for the heavy ion plant which contributes to a smaller cavity. Lastly, the transport channel dramatically reduces the number and size of first wall penetrations—two 2-cm diameter openings compared to sixty ~20 cm diameter openings for the laser. The reactor plant cost reflects these differences. It should be noted that the cost difference could be even larger for the two systems. Cavity vapor pressure limits correspond to a radius of ~3.8 m for the heavy ion system; however, study resources did not permit a detailed assessment of the effects of the higher yield and blanket power density on a cavity with this small a radius.

A final comment involves the cost estimate of the target factory for the two systems. The laser design employs direct drive targets which are envisioned as simple hydrocarbon shells filled with frozen DT. The heavy ion targets are indirect drive with a simple shell suspended in a radiation case that has energy conversion regions at the opposing ends. As discussed in Section 6.4, the costs for the heavy ion target factory are higher than those for the laser system, but only by ~6.4%. This is partially due to the lower repetition rate for the higher yield heavy ion system, 3.54 pps compared to 5.65 pps for the laser. The heavy ion target factory thus manufactures only 63% as many targets per year. However, it should also be noted that concerns about the complexity and cost of heavy ion targets do not appear warranted. The only major difference is the radiation hohlraum. Suspension of the capsule inside the case is a concern, but analyses indicate that a low cost solution might be possible.

Cost sensitivity studies for the Prometheus designs were conducted to assess the relative impact to factors of concern to utilities. Figure 6.13-2 shows how the design benefits from a level of safety assurance of 1 for both the laser and heavy-ion systems. Guidelines for assessing the cost impact of LSA were developed by Holdren[5] and are summarized in Table 6.13-2. The bases for the economic guidelines are discussed in Section 3.5, Economic Guidelines. The factors shown in Table 6.13-2 were used to calculate the data displayed in Figure 6.13-2. This shows that the COE would be ~13% higher for LSA = 4 which is typical of present day nuclear fission plants. However, the story is incomplete because this type of plant would likely employ cheaper cavity and shield structural materials than those used for the Prometheus design.

Table 6.13-2. Indirect Cost Factors for Different LSA Values

	LSA 1	LSA 2	LSA 3	LSA 4
Construction Services	0.113	0.120	0.128	0.151
Home Office Engineering	0.052	0.052	0.052	0.052
Field Office Engineering	0.052	0.060	0.064	0.087
Owners Cost	0.150	0.150	0.150	0.150
Process Contingency	0	0	0	0
Project Contingency	0.1465	0.173	0.184	0.195
Plant O&M Factor	0.700	0.850	0.952	1.0
Decommissioning (M$/yr)	4.56	9.12	13.68	18.23

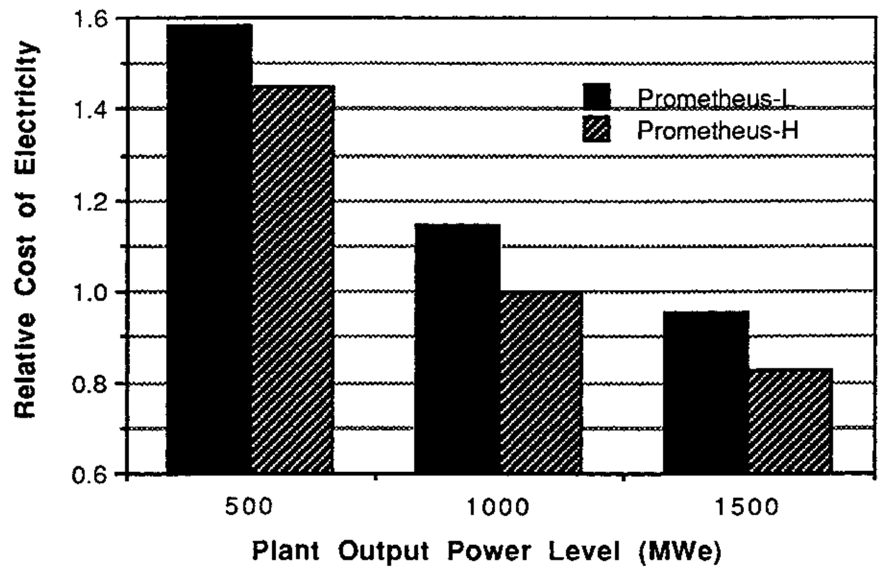

Figure 6.13-2. Change in Prometheus COE for Different LSA Assumptions

Economy of scale is another important economic characteristic of a central station power plant . The projected variation in COE for the Prometheus plants with electric power output is depicted in Figure 6.13-3. This figure indicates the economies of scale for large power plants, however it is interesting to note that the heavy ion system is still significantly better than the laser (~8.9%) at 500 MWe, even though the driver cost is virtually unchanged. This is because of the significant ηG advantage for the heavy ion system. This is a more favorable result than historically expected for the heavy-ion driver and it is a direct outgrowth of the potential cost savings afforded by the innovative, single beam configuration.

The driver output energy (hence overall driver size, ion energy, etc.) was not reoptimized at each different power level for this study, only the pulse repetition rate was varied. Slightly different results might be expected for an optimized design. However, the conclusions are not expected to change much if the same gain curves are employed. This brings up an important point – there may be more attractive target designs for lower output power plants. Such targets would provide moderate gain at low (1-2 MJ) drive energies that would reduce the required size (and cost) of the driver, possibly leading to more cost attractive 100-500 MWe power plants. This is the subject of one of the critical issues identified by the present study and it is discussed in Section 5.

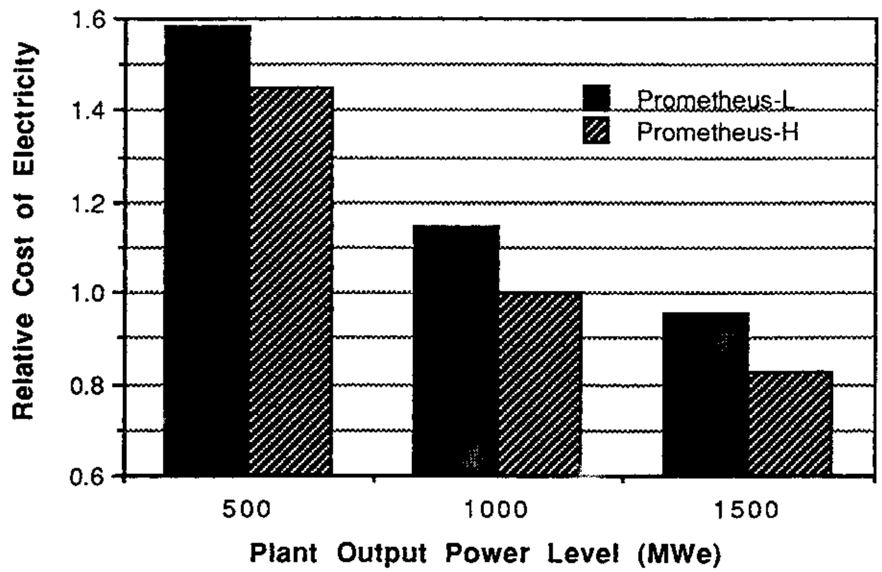

Figure 6.13-3. Change in Prometheus COE for Different Plant Electric Power Outputs

The conclusions of the economic studies are that IFE power plants can be designed and constructed to have reasonable economics to effectively compete with other energy sources. The innovative approaches used in the driver designs have lowered costs for both the laser and the heavy ion driven power plants so that they are now more than competitive with comparable MFE designs. Indeed, if the using community, be it utilities or owners/operators, can effectively use the larger sized plants, then IFE will be a clear winner from the standpoint of more affordable power. The flexibility of the separation of the driver from the reactor chamber offers several advantages including the use of very low activation materials. The low Level of Safety Assurance is also a significant economic benefit. The economic analyses affirm that IFE should be further developed as a future energy source.

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