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**PRELIMINARY SAFETY AND LICENSING
INFORMATION DOCUMENT
SUBSEA NUCLEAR POWER
GENERATING STATIONS FOR
OFFSHORE OIL PRODUCTION OPERATIONS**

ERDA Research and Development Report

*Prepared for the United States
Energy Research and Development Administration
Division of Nuclear Research and Applications
under Contract Number E(04-3)-701*

Rockwell International

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ABSTRACT

This document presents preliminary safety and licensing information relating to the manufacture, installation, and operation of small subsea nuclear generating stations in remote offshore locations, and the associated on-shore base site activities of reactor refueling and system refurbishment at 4-year intervals.

The purpose of the proposed facilities is to provide a subsea power capability for offshore oil production operations in areas where surface conditions, water depths, or other environmental constraints may preclude the use of surface-mounted power sources. This will provide the capability to achieve the maximum recovery of petroleum resources from the U.S. frontier areas.

Summary descriptions of a typical subsea nuclear generating station and a typical base site facility are presented. General safety and licensing considerations are discussed, and a representative schedule of licensing activities is outlined.

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I. PURPOSE OF PROPOSED FACILITIES

The proposed facilities include one or more small (3,000 to 15,000 kWe) nuclear power systems, designed for installation on the sea floor to provide power for offshore oil and gas production operations, and a base site which will be utilized for the final assembly of the nuclear power systems, the initial loading of the reactor fuel, and for subsequent refueling and refurbishment activities, which are scheduled to occur at 4-year intervals.

The purpose of the proposed facilities is to provide a subsea power capability for offshore production operations in areas where surface conditions, water depths, or other environmental constraints may preclude the use of surface-mounted power sources. Power systems of this type may be a necessary component of subsea oil and gas production systems that will play an important role in recovering petroleum resources located in the deeper parts of the Outer Continental Shelf and in some of the Arctic offshore regions.

The achievement of substantial recovery of the resources in these frontier areas is of considerable importance to future U.S. energy supply. Although a number of development options to deal with the overall energy problem are being pursued, the current projection is that, even with extensive technological success in all development options, the U.S. will still be faced with a serious oil supply problem in the coming decades.

The role of oil and gas is so pervasive in the nation's energy economy that the highest priority must be assigned to locating and recovering more oil and gas from domestic sources. Only two areas have significant potential for increased petroleum production: the Federal Outer Continental Shelf, and the northern regions of Alaska. In other areas (onshore and state offshore regions), most of the proved reserves supporting today's production will be substantially reduced by 1985.

The availability of the proposed subsea nuclear power facilities will contribute to the achievement of maximum recovery of petroleum resources from the U.S. frontier areas.

Figure 1. 3000-kWe Subsea Nuclear Power Plant

Figure 2. Installation of Subsea Nuclear Power Plant

II. SUMMARY DESCRIPTION OF FACILITIES

It is anticipated that this application will involve the installation of a number of subsea generating stations, perhaps between 10 and 50, with power capabilities ranging from 3,000 to 15,000 kWe, in the frontier areas of off-shore oil development. These areas will generally be the Alaskan Arctic off-shore regions, where ice is expected to be the principal constraint to conventional development methods that rely on surface support facilities, and the more remote (20 to 200 miles offshore), deeper (600 to 3,000 ft) regions of the outer continental shelf (OCS).

To provide the onshore facilities needed for refueling and refurbishing the subsea stations, from one to three base sites will be required, potentially in Alaska, on the West Coast, and on the East Coast of the U.S.

A. NUCLEAR GENERATING STATION

The Subsea Nuclear Power Generating Station described here is designed to produce 3,000 kW of electrical power. The overall system lifetime is 20 years, with refueling and refurbishment occurring on a 4-year cycle.

1. Power System Description

The power system configuration is shown in Figure 1. The major subsystems are: (1) a vertical pressure vessel containing the ZrH reactor heat source, primary coolant system, and boiler, (2) a horizontal pressure vessel containing the organic Rankine power conversion system, the electrical generator, and control panels, and (3) the waste heat exchanger, a tube bundle located between the two pressure vessels. The system is installed on a mounting structure designed for loading on a barge. The barge is designed to serve also as a seafloor support structure. The barge is ~48 by 84 ft in size, and is designed to be towed to the offshore production site.

At the site, the bottom support structure is first lowered to the seafloor, positioned, and secured to the ocean bottom. The power system, on its support structure, is then lowered onto the bottom mounting, using a combination

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Figure 3. Power System Schematic

Figure 4. Reactor Module

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of downhaul lines and controlled buoyancy (Figure 2). Once securely in place on the seafloor mounting, additional negative buoyancy is added by flooding additional ballast tanks to provide a large negative buoyance. Power cables are laid to the pumping or production station, and control cables are run to the production supervisory control center.

The system may be started up either from the control panel in the generator module or from the control panel in the production control center. The design life of the reactor core is 4 years. Consequently, at 4-year intervals, a replacement power system is brought to the production site and interchanged with the spent system, the bottom mounting remaining in place on the seafloor. The spent system is towed to a refurbishment center, where the reactor core is replaced and other parts of the system are inspected, and refurbished or replaced, as required.

Figure 3 is a simplified schematic of the nuclear power system. Thermal energy is generated in the zirconium hydride reactor, and is transferred to the organic boiler via a liquid metal coolant loop. In the boiler, the liquid metal coolant, NaK (a eutectic mixture of 22% sodium and 78% potassium), transfers the thermal energy to the toluene (CP-25) working fluid of the organic Rankine power conversion system. The toluene is vaporized, and the vapor expands through the turbine which drives the electrical generator, producing 3300 kWe gross. The toluene vapor from the turbine exhausts to the regenerator, then to a condenser. Waste heat from the power system is then dissipated to the seawater through a waste heat rejection system. The waste heat rejection system also uses an organic fluid mixture (e.g., 50% ethylene glycol, 50% water) having a freezing point below -20°C . The waste heat rejection system transfers heat to the seawater by natural convection around the heat exchanger tubes.

2. Reactor Module

The reactor module (Figure 4) contains the reactor, reactor control rods and drives, the primary coolant piping, primary coolant pump, boiler, an expansion tank to take up volume changes as the primary coolant temperature changes, and the radiation shield. The pressure vessel is 22 ft long, and has a maximum diameter of 10 ft.

The heat source is a zirconium hydride reactor. This reactor concept was selected, early in the Space Nuclear Power Program, as the most suitable reactor type to provide a small, compact nuclear heat source for long-term, unattended operation in space. It has been under development since 1956. Based on past program experience, the zirconium hydride reactor is well suited to provide the 4-year unattended life required by the subsea system.

The reactor core contains 19 fuel assemblies, arranged in a hexagonal array. Each fuel assembly contains 72 wire-wrapped fuel elements, located between an inner and outer stainless steel hexagonal tube. A B_4C control rod is located inside the inner hexagonal tube.

Seven control rod drive motors drive the 19 control rods. The 18 outer control rods are ganged in sets of 3, so that 6 motors each drive 3 rods; the seventh motor drives only the central control rod. A rack and pinion drive, utilizing commercially available components, has been tentatively selected.

The reactor core is surrounded by a beryllium reflector, a stainless steel thermal shield, and a serpentine concrete neutron shield. NaK, at 600°F, enters the reactor core from two inlet lines at the bottom, and exits via two outlet lines at 800°F. The NaK then enters a tube-and-shell heat exchanger, where the higher-pressure (600 psia) organic fluid (toluene) is contained in the tubes. The NaK is pumped by an ac helical induction pump. An expansion compensator is used to accommodate the change in NaK volume, as the NaK is heated to an average temperature of 700°F, and to pressurize the primary system to 2 atm abs.

The use of a liquid metal reactor coolant allows use of an ac helical induction pump in the reactor primary system. This pump operates entirely by electromagnetic effects. It has no moving parts, bearings, or seals, and should therefore provide high reliability for long-term operation. This also permits the entire reactor primary coolant system to be an all-welded, hermetically sealed system — an important safety and reliability feature.

The concrete radiation shield has been sized, based on several considerations: (1) to limit the radioactivity induced in the surrounding seawater to levels permitted by current regulations (10 CFR 20), (2) to minimize neutron activation of the pressure vessel, so that replacement and refurbishment of the system will not require undue precautions with respect to radioactivity levels, and (3) to limit the radiation intensities in the generator module, at full-power operating conditions, to levels that permit entry into that module by workmen for reasonable periods for monitoring or maintenance activities.

The reactor module pressure vessel is made of ASTM A516, and has a thickness and frame spacing determined by the desired operating depth. It is flanged at one end for ease of refurbishment. Also, a small flanged nozzle at the end is provided for a pipe chase containing the organic piping, and control and instrumentation cables.

3. Generator Module

The generator module contains the turbine, electrical generator, regenerator, condenser, hot well, organic pumps, waste heat coolant pump, electrical and control panels, and storage batteries for startup power. Figure 5 shows a typical generator module design. The pressure vessel is 10 ft in diameter and 28 ft long. The access hatch is designed to mate with the Lockheed subsea service module. Electrical power can be delivered either to cables which exit the generator module through a Lockheed bull-nose port pressure seal or to feed-throughs which penetrate the module wall to an electrical connector fitting.

The organic working fluid (toluene) enters the turbine at 700°F and 500 psia. The vapor expands to 1.75 psia and 466°F, then flows through the regenerator, the condenser, and into the hot well. From the hot well, the condensed organic liquid is pumped back through the regenerator for preheating, then on to the boiler in the reactor module.

The turbine is a single-stage, supersonic, axial-impulse, full-admission design, rotating at ~8000 rpm. It has a diameter of 34 in. and a blade height of 5.2 in. Two types of electrical generators are being considered for this

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Figure 5. Generator Module

system. One is a 400-Hz generator which can be directly coupled to the turbine shaft, and would thus provide the advantage of making the entire combined rotating unit (CRU) — turbine, generator, and main organic pump — a hermetically sealed unit operating on a single shaft. A generator of this type would be of minimum size and weight, but would require some development, and would be more costly in production than a more standard 60-Hz unit. Also, a frequency converter would be required to deliver the 60-Hz power required for conventional oil production equipment.

The other generator option is a 60-Hz unit of more standard design and lower in cost. In this case, however, a gear box is needed between the turbine and the generator, and the larger size of the 60-Hz unit also requires a larger pressure vessel. Further trade studies will be performed to evaluate these generator options.

Normally, both the generator module and reactor module are operated in a nitrogen-rich atmosphere, in order to minimize the possibility of a fire.

The toluene fluid is condensed in the condenser, and the waste heat transferred to another organic system, the waste heat rejection system. The waste heat rejection system is designed to dissipate the waste heat from the organic Rankine cycle to the ocean. The key component in this system is the waste heat exchanger, which is mounted on the deck between the reactor and the generator pressure vessels. The heat exchanger dissipates its heat via natural convection and ocean currents. A self-contained, natural convection heat exchanger was selected, in order to avoid the problems associated with potential plugging of ocean intake screens with silt, debris, small fish, etc. Preliminary design indicates that ocean currents of 1/2 kn are sufficient.

The coolant used in this system is a mixture of water and organic, so that the heat transfer properties are maximized but the possibility of freezing is minimized. All the piping and components in the waste heat exchanger are designed to withstand an internal pressure equivalent to the external head of sea water, so that the integrity of the system internal to the pressure vessel is maintained.

A more complete description of this nuclear power system, and the major components in the reactor and generator modules, is presented in AI-ERDA-13194, "Conceptual Design Description, 3000-kWe Subsea Nuclear Power Plant" (Reference 1).

B. BASE SITE

The base site will provide facilities for the following activities:

- 1) Final assembly of the reactor module, the generator module, and the waste heat exchanger, and installation of these systems on the barge
- 2) Receipt, inspection, and storage of new fuel assemblies
- 3) Loading fuel assemblies into the reactor core vessel
- 4) Final closure of reactor vessel
- 5) Loading NaK into the reactor primary system
- 6) Final closures of reactor module
- 7) Performing dry and wet low-power criticality tests and final checkout of reactor systems
- 8) Install main pipe chase containing instrumentation and control cables and organic piping; complete connections and closures in reactor and generator modules
- 9) Fill power conversion system and waste heat exchanger with organic fluids
- 10) Perform final system checkouts
- 11) Receive spent barge-mounted nuclear generating stations (after 4-years operation at offshore production site)

- 12) Dry-dock barge, visually inspect
- 13) Drain power conversion system and waste heat exchanger of organic fluids, back-flush, clean, and seal; store or dispose of fluids
- 14) Disconnect instrumentation and control cables, organic piping, and pipe chase
- 15) Remove reactor module head
- 16) Drain NaK from primary coolant system, flush, and backfill with nitrogen
- 17) Remove top shield plug and core assembly into shielding cask
- 18) Remove spent fuel assemblies, move to spent fuel storage, and prepare for shipment
- 19) Inspect and test reactor vessel, pressure vessel, and head
- 20) Refurbish reactor system components
- 21) Install new top shield plug and core assembly
- 22) Refurbish power conversion system and waste heat exchanger
- 23) Final assembly of reactor module, generator module, and waste heat exchanger on the barge.

A more detailed list of base site activities is contained in AI Techninal Information Document TI-652-600-001, dated September 5, 1976 (Reference 2).

A layout of the general arrangement of a base site facility is shown in Figure 6.

- LEGEND**
- 1. SUB STATION
 - 2. SECURITY
 - 3. NEW FUEL CONTROL
 - 4. OPERATING PERSONNEL CONTROL
 - 5. NEW FUEL STORAGE VAULT
 - 6. RADIOACTIVE WASTE & SPENT FUEL VAULT
 - 7. WHARF SIDE WORK AREA
 - 8. NEW FUEL & CORE ASSEMBLY
 - 9. DRY DOCK
 - 10. CONTROLLED ACCESS MACHINE SHOP
 - 11. MACHINE SHOP
 - 12. ADMINISTRATIVE AREA
 - 13. GENERAL REPAIR AREA
 - 14. DOCK
 - 15. HOT CELL
 - 16. OPERATING GALLERY
 - 17. PARKING

BASE SITE FACILITY

GRAPHIC SCALE
0 50 FT 100 FT
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Figure 6. Base Site Facility

C. MANUFACTURING FACILITIES (California)

The manufacture of components and subassemblies, including fuel assemblies, for the subsea power generating stations will be performed in facilities within the Atomics International complexes in Canoga Park and Santa Susana, California (Figures 7 and 8). The AI headquarters complex in Canoga Park comprises five buildings, housing some 1200 management, engineering, and technical personnel, laboratory facilities, and a 140,000 ft² shop manufacturing area. Laboratories are equipped for chemical analysis, evaluation, and quality control of materials, and for development and testing of components. An integrated fabrication, inspection, and quality assurance organization is centralized in the manufacturing facility, which normally utilizes a stable labor force. AI currently holds ASME "N," "NPT," and "U" certificates of authorization to manufacture nuclear pressure vessels, steam generators, and piping components.

The Nuclear Development Field Laboratory is a major field test area for large engineering and nuclear development and test projects. It occupies a 290-acre site in the nearby Santa Susana Mountains. The laboratory includes a plutonium fuel facility, an extensive hot cell facility, and numerous facilities for component and system development and test, system assembly, and nuclear test.

These facilities have been utilized for the past two decades for a broad range of nuclear research, development, manufacturing, and fuel fabrication programs. AI currently holds a Broad Radioactive Materials License, covering fabrication of several types of fuel elements for nuclear reactors. Over the past 20 years, about 30,000 fuel elements, utilizing some 100,000 kg of enriched fuel, have been fabricated in these facilities.

It is not expected that the proposed subsea plant manufacturing activities will introduce safety or environmental impacts of a nature or magnitude substantially different from those associated with the on-going and continuing operations in these facilities.

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Figure 7. Atomics International Headquarters, Canoga Park, California

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Figure 8. Atomics International Nuclear Development Field Laboratory

III. GENERAL LICENSING CONSIDERATION

A. LEGAL FRAMEWORK^(3,4)

The NRC's licensing and regulatory authority is derived mainly from two statutes: the Atomic Energy Act of 1954, as amended, and the National Environmental Policy Act of 1969 (NEPA). Under the Atomic Energy Act, the primary substantive considerations in licensing relate to the public health and safety and the common defense and security. NEPA is the basic source of the Commission's authority with regard to environmental matters, as well as the basis for several procedural requirements regarding the licensing of nuclear plants, including the requirement that a detailed environmental impact statement be prepared.

Jurisdiction over the sea is split into several categories, some of which overlap. Under the various Geneva Conventions to which the United States is a party, the ocean is divided into these jurisdictions: (1) the territorial sea, (2) the contiguous zone, (3) the high seas, and (4) the continental shelf.⁽⁵⁾

The territorial sea is that marginal belt of the sea which is incorporated under a nation's sovereignty. The only respect in which this sovereignty is not complete is that foreign vessels have the right of innocent passage into this belt. Although the Convention on the Territorial Sea and the Contiguous Zone does not specify the width of the territorial sea, the 12 nautical mile limit is becoming universal. The United States, however, does not subscribe to this international trend, and claims only a 3 nautical mile limit.

The contiguous zone, extending 12 nautical miles from the coastal state, is a zone in which the coastal state may implement regulations concerning customs, immigration, fiscal, and sanitary policy. The United States considers this to be an exclusive fisheries zone, as well.

Overlapping the contiguous zone are the high seas, consisting of world oceans excluding internal waters and the territorial sea. A state has these rights under the Convention on the High Seas: (1) freedom of navigation, (2) freedom of fishing, (3) freedom to lay submarine cables and pipelines, and (4) freedom to fly over the high seas. To go with these rights are certain obligations — most important for the purposes of this study are the obligations of a

state to take safety measures for ships under its flag, and to comply with measures established under international authority to prevent radioactive pollution of the seas and airspace above.

Finally, there is the continental shelf. The Convention on the Continental Shelf defines the continental shelf this way:

"For the purpose of these articles, the term 'continental shelf' is used as referring (a) to the seabed and subsoil of the submarine areas adjacent to the coast but outside the area of the territorial sea, to a depth of 200 meters or, beyond that limit, to where the depth of superjacent waters admits of the exploitation of the natural resources of the said areas; (b) to the seabed and subsoil of similar submarine areas adjacent to the coasts of islands."(6)

For the purposes of this discussion, the "depth of exploitation" criterion determines the United States' continental shelf.

Jurisdiction over the continental shelf entails exploration and exploitation of its natural resources. To the extent that subsea nuclear generating stations are utilized only for purposes of exploration and exploitation of natural resources, then the Convention on the Continental Shelf establishes United States jurisdiction over such stations, and provides the basis for licensing jurisdiction, under the Atomic Energy Act condition that:

"No license under this section may be given to any person for activities which are not under or within the jurisdiction of the United States, except for the export of production or utilization facilities" (7)

Rights over the continental shelf do not affect the legal status of high seas and airspace above the shelf. The Convention also does not authorize nonextractive uses of the shelf.

As far as United States law is concerned, the most important statutes are the Outer Continental Shelf Lands Act⁽⁸⁾ and the Submerged Lands Act.⁽⁹⁾ The Outer Continental Shelf Lands Act delineates United States authority over the continental shelf. It specifies leasing procedures for mineral resource exploitation, and extends Army Corp of Engineers authority to fixed structures and

artificial islands exploiting the shelf's natural resources. Like the Convention on the Continental Shelf, the Outer Continental Shelf Lands Act does not authorize leasing for nonextractive purposes.

The Submerged Lands Act was passed by Congress in response to a Supreme Court decision⁽¹⁰⁾ asserting that the submerged land within the 3-mile limit belonged to the United States, and not the states. Congress thought this legislation gave this submerged land back to the states, but reserved to itself special rights concerning navigation, flood control, and the production of power.⁽¹¹⁾ The controversy did not end there, however. Despite numerous Supreme Court decisions,⁽¹²⁾ the precise boundary between federal and state land has yet to be determined.

There is a legal advantage to having a subsea nuclear power plant far from the continent. Namely, if there happens to be a nuclear accident, the chances of injuring or destroying property and lives on shore are reduced, and thus, so are damage payments. In addition, adverse impact on recreation and aesthetics is less likely. All in all, public opposition would probably be less. Finally, a site beyond the 12-mile limit might be outside the jurisdiction of some governmental agencies.

While other federal and state agencies may play a significant role in connection with offshore power plants, the emphasis here is on the NRC regulatory process, and on the legal framework which has been devised for those plants and set forth in current NRC regulations.

The NRC's licensing regime for nuclear power reactors has, in the past, been comprised of two basic stages — the construction permit stage, and the operating license stage. The construction permit review is based essentially on the final design. A public hearing is mandatory at the construction permit stage, even if the grant of the permit is unopposed; at the operating license stage, only an offer of a hearing is required. A safety review by the Commission's Advisory Committee on Reactor Safeguards is performed at both of these stages.

The rules and regulations relating to the facility licensing process have, in general, been predicated on the assumption that the facility will be assembled

and constructed on the site at which it is to be operated. In those circumstances, both facility design and site-related issues could be considered in the initial construction permit stage of the licensing process.

However, under the Atomic Energy Act, a license may be sought and issued authorizing the manufacture of facilities, but not their construction and installation at the sites on which the facilities are to be operated.

Under NRC regulations, an applicant for a manufacturing license would be required to include in his application, in addition to design information, the postulated site parameters. In general, the design information required is similar in scope and depth to that required in an application for a construction permit. Since the site or sites at which the reactors are to be operated may not be known at the time the application for a manufacturing license is submitted, and, in any event, need not be identified in that application, there is a provision that any required information or analyses relating to site matters are to be predicated on the postulated site parameters.

The issuance of a license to manufacture nuclear power reactors is considered to be a "major Federal action significantly affecting the quality of the human environment," as that term is used in NEPA. The applicant would be required to submit with his application an environmental report, as applicants for construction permits are now required to do. The report will cover the environmental impact of the manufacturing activities, and, in general terms, the construction and operation of the reactors at a hypothetical site or sites having characteristics that fall within the postulated site parameters. The detailed environmental impact statements prepared by the Commission will be similarly directed. The purpose here is to assure that environmental impact is considered as early in the licensing process as is feasible.

The application will be reviewed by the Commission's Advisory Committee on Reactor Safeguards, and a public hearing on the application will be mandatory -- with essentially the same procedures as those now applied at construction permit hearings.

One interesting feature of this framework is the requirement that the holder of a manufacturing license submit the final design of the reactors as soon as

that design has been completed. This is in keeping with one of the basic purposes of the framework, which is to facilitate and encourage early decisions in the overall review process. Under the two-stage process which has been the norm up to now, there is no requirement for the owner-operator applicant to submit the final design until the operating license stage. The submittal of the final design by the holder of the manufacturing license is to be in the form of an application for amendment of the manufacturing license.

In the construction permit proceeding, the focus would be on the specific site chosen by the owner-operator applicant, including whether that site fell within the postulated site parameters specified at the manufacturing license stage. In cases where a manufacturing license is sought, no construction permit would be issued to the owner-operator until the manufacturing license had first been issued to the manufacturer. The NEPA review process now performed at the construction permit stage would, of course, be conducted, except that the environmental report required of the owner-operator applicant would not be expected to duplicate matters covered in the environmental report submitted in connection with the application for a manufacturing license.

The reactors may not be transported from the manufacturing site to the site where they will be operated until the construction permit has been issued. Furthermore, the reactors may not be removed from the manufacturing site until the Commission has approved the final design.

While the procedural aspects of the operating license stage are largely unchanged, it is important to note that the final design would have been approved before that stage. This points up another major purpose of the regulations — the elimination of unnecessary duplication in the review process. Matters resolved at any stage of the licensing process would, in general, not have to be reconsidered. The Commission could, however, reopen matters previously resolved, on the basis of significant new information that substantially affects the conclusion(s) reached at the earlier stage.

This, then, is the regime proposed for the fabrication, assembly, and operation of nuclear power plants which will be operated in an ocean environment. Even if the reactor were to be located outside the "3-mile limit" of the

United States, this regime could be followed, by virtue of the "in personam" jurisdiction which the United States has over the activities of its citizens. The question of the extraterritorial extent of the Commission's facility licensing jurisdiction was considered in connection with the licensing of the N.S. Savannah. There, it was concluded that the AEC had general licensing authority over all nuclear-powered vessels documented under the laws of the United States. While the conclusion turned, in part, on national jurisdiction over merchant vessels (which are traditionally considered to be national territory), the basic reasoning applied (i.e., that "jurisdiction", as used in the licensing sections of the Atomic Energy Act, can be construed to mean the jurisdiction of the sovereign over its subjects) is also applicable to the barge-mounted reactor.

The question whether an ocean-based reactor, located beyond the territorial limits of the United States (3-miles, under current doctrine), could be indemnified under the "Price-Anderson Act" is a related, but different, question. The "Price-Anderson Act" amended the Atomic Energy Act to provide for a system whereunder the public will be protected, in the event of certain nuclear incidents, by private insurance and overriding Government indemnity totaling \$560,000,000. Under "Price-Anderson," indemnity for licensees seems quite clearly to be limited to nuclear incidents "within the United States" in a geographical sense. "Nuclear incident" is defined as an occurrence within the United States, with a proviso to the effect that the term includes an occurrence outside the United States in the case of the N.S. Savannah, and in the case of a facility owned by or used under contract with the Commission (ERDA). The net effect of all this is that "Price-Anderson" indemnification would not appear to be available, under current law, for licensed nuclear power reactors which are located beyond the "3-mile limit."

As noted at the beginning of this section, this discussion outlines the NRC facility licensing process, and the related legal framework for offshore nuclear plants. It does not fully cover the legal implications of this subject.

B. PRINCIPAL LICENSING QUESTIONS

It is anticipated that the principal licensing questions will arise as a result of the proposed operation of the subsea nuclear generating stations in the unattended mode. Unattended operation of a reactor power system is a new and untried concept, with respect to licensing and regulatory review. Particularly with regard to nuclear safety and physical protection of the reactor facility against industrial sabotage, deviations from current licensing requirements for land-based reactor facilities, or new approaches to the corresponding requirements for offshore subsea reactors, may be needed.

No unusual licensing issues are anticipated, in connection with the base site facility or the manufacturing facilities.

1. Nuclear Safety

The basic nuclear safety issue with any reactor facility is the potential hazard to the public from the release of radioactivity. The topics included in the Environmental Report Outline (Appendix A), and in the Preliminary Safety Analysis Report Outline (Appendix B), provide guidelines for the generation of the information required as an adequate basis for the issuance of a license.

Information of this nature will be generated for the subsea nuclear generating station, and for the base site, during the preliminary design and final design activities. Some preliminary comparisons, of a general nature, have been made between the safety aspects of the subsea reactor and the safety aspects of a typical commercial water reactor utilized in nuclear electric stations now in operation or under construction in the United States. These preliminary comparisons can serve to put the subsea reactor safety questions into perspective.

A study of the comparative risks and consequences between 1150-MWe Floating Nuclear Plants (FNP), as proposed by Westinghouse Offshore Power Systems, and similar land-based nuclear plants, associated with accidental release of radioactive material has been made by the staff of the Division of Site Safety and Environmental Analysis of NRC.⁽¹³⁾

The staff's environmental impact statements on light water reactors conclude that the risk associated with accidental radiation exposure is very low. The validity of this conclusion, in the case of Floating Nuclear Plants (FNP's), depends on three principal factors: (1) any significant differences in the probability of an accidental release, (2) any significant differences in the magnitude of releases associated with various accidents, and (3) any significant differences associated with the pathways by which released materials may reach man.

Based on a review of the FNP design, a spectrum of potential accidents was selected for purposes of the NRC study. The accidents ranged from minor operational incidents to very improbable core-melt events. Consequences to man for all accidents were estimated in terms of radiation doses from drinking water ingestion, consumption of aquatic foods, and direct exposure from swimming and beach activities. For the core-melt events, the principal assessments were directed to radiation dose to man and undesirable modifications of ecosystems.

The report contains site descriptions, representative accident scenarios, related source terms, hydrologic dispersion and radionuclide transport models, radiation dose models, descriptions of ecological consequences, and a discussion of methods for mitigating adverse consequences.

Hydrologic dispersion parameters, potential food chain interactions, human dose pathways, and interdiction methods for accidental releases of radioactivity from FNP's are different from those of land-based plants. Therefore, judgment of the relative risks of the two types of plants depends upon the degree to which the consequences of accidents are comparable. If the consequences are equivalent to or less than those for land-based plants, the risks of the proposed FNP's would be no greater than those associated with licensed land-based PWR nuclear power plants of a similar type.

The consequences of accidental releases of radioactivity to the atmosphere from FNP's are within the envelope of atmospheric impacts from land-based plants, and are discussed in the staff's Safety Evaluation Report for the FNP. The consequences of washout and deposition of airborne releases into surface waters were judged to be negligible, in comparison with the direct releases, and thus were not calculated.

Consequences of accidental releases to the liquid pathway, ranging from normal operational releases to those that might result from an improvable core-melt accident (in excess of design basis), were evaluated for typical land-based and water-based sites. In general, for accidents involving design basis events, the resulting consequences at ocean-based FNP's are about one order of magnitude lower in estimated impact to man than those from comparable land-based plant accidents.

Core-melt events in light water reactors of this type are likely to be accompanied by containment failure due to overpressurization. This is not expected to vary, whether they are land-based or FNP's. Releases via the liquid pathway are expected to be a slow leaching of radioactive material from core debris, subsequent to containment melt-through. The resulting liquid pathway consequences for FNP's are evaluated to be comparable to, and within the envelope of, the consequences assessed for land-based plants. In the staff's judgment, the consequences of such releases will not be significant, compared to potential airborne releases from the same event or scenario.

The relevance of these study results to the small subsea nuclear generating stations requires further comments regarding the principal validity factors stated previously:

1) Any significant differences in the probability of an accidental release.

The subsea nuclear plant will be subject to safety design, quality assurance, and operational standards comparable to those applied to light water reactors. A lower probability of off-normal or accident conditions may be achieved in the subsea plant, due to the relative simplicity of the system and to certain inherent safety features not present in water reactors.

The primary (radioactive) coolant system of the subsea reactor is a completely self-contained, all-welded circuit. There are no auxiliary systems, valves, seals, etc, with attendant possibilities for accidental radioactivity release. The primary coolant piping is a compact, low-pressure configuration; this diminishes problems associated with pipe stresses and the adequacy of piping supports.

Further, since an inert nitrogen atmosphere is maintained continuously in the reactor module of the subsea plant, it is not dependent on successful activation of an additional safety system, such as a hydrogen-oxygen recombiner, to prevent the possibility of explosion of these gases, as are water reactors.

With respect to inherent safety features, the normal release of gaseous fission products from the UZrH fuel of the subsea reactor is $< 1\%$ of the total inventory; in the UO_2 fuel of light water reactors, $\sim 95\%$ of the gaseous fission products are released from the fuel. Also, the subsea reactor can lose reactivity, and thereby limit power and temperature excursions, by loss of hydrogen from the fuel moderator. Similar reactivity losses in light water reactors are related to loss of the coolant by vaporization or expulsion from the core, resulting in a higher probability of fuel melting.

- 2) Any significant differences in the magnitude of releases associated with various accidents. The reactor core is the major source of radioactivity generated by nuclear plant operation. The total inventory of radioactivity in the core of the 3000-kWe (10-MWt) ZrH subsea reactor, for a given period of operation at full design power, is ~ 350 times smaller than the corresponding radioactivity inventory in the core of a 1150-MWe (3425-MWt) light water reactor.
- 3) Any significant differences associated with the pathways by which released materials may reach man. The NRC study concludes that, for 1150-MWe Floating Nuclear Plants located 2000 m (1.25 miles) offshore, the consequences of releases to the atmosphere are within the envelope of atmospheric impacts from land-based plants, for which the risk is very low; and the consequences of releases to the liquid pathway are about one order of magnitude lower in estimated impact to man than those from comparable land-based plant accidents. The 3-MWe subsea nuclear plants are further removed, in terms of pathways to man, by immersion in water to depths of 600 to 3000 ft, and by distances of 20 to 200 miles offshore. In many cases, subsea plant locations will be in the Alaskan arctic regions.

A point of reference, in this connection, is that the volume of air required to dilute a given amount of Sr^{90} to the maximum permissible concentration, $\text{MPC}_{(\text{air})}$ (general public) is 10,000 times the volume of water required to dilute the same amount of Sr^{90} to $\text{MPC}_{(\text{water})}$ (general public).

2. Protection Against Industrial Sabotage

Atomic Energy Commission Regulation 10 CFR, Part 50, "Licensing of Production and Utilization Facilities," requires each licensee authorized to operate a nuclear reactor to provide appropriate protection against acts of industrial sabotage which could lead to a threat to the health and safety of the public. Criteria for industrial security programs to provide such protection are contained in American National Standards Institute (ANSI) Standard N18.17, dated March 23, 1973, as supplemented by USAEC Regulatory Guide 1.17, June 1973.

From ANSI N18.17:

"A suitably comprehensive industrial security program comprises those measures taken to thwart attempted sabotage. Means to accomplish this purpose include measures to control entry to the plant site or portions of the facility and to deter or discourage penetration by unauthorized persons; to detect such penetrations in the event they occur; to apprehend in a timely manner unauthorized persons, or authorized persons acting in a manner constituting a threat of sabotage; and to provide for appropriate authorities to take custody of violators. The means by which such measures are implemented generally fall into five major subject areas, each of which is discussed in subsequent sections of this standard: (1) physical and administrative control of access to the plant site or portions thereof; (2) selecting and retaining reliable personnel and for detecting aberrant behavior of authorized personnel; (3) monitoring the status of vital equipment; (4) augmenting security in the event of actual or potential threats to plant security; and (5) design features of the plant specifically provided for security purposes and design features (such as remoteness of the site, separation and redundancy of vital equipment, shielding of radiation sources or containment structures) provided for reasons that may or may not be associated with security but which reduce the vulnerability of the plant to sabotage attempts."

The principal supplements of USAEC Regulatory Guide 1.17 relating to the proposed unmanned operating mode of the subsea nuclear generating stations are contained in the following excerpt:

"C. REGULATORY POSITION

"The requirements and recommendations contained in the proposed ANSI Standard N18.17, 'Industrial Security for Nuclear Power Plants,' dated March 23, 1973, are generally acceptable and, with due consideration for the unique characteristics of the plant and its owner organization, provide an adequate basis for a physical security plan for the protection of nuclear power plants against industrial sabotage, as supplemented by the following:

1. Security Systems

- a. The plant security forces should have onsite, armed, and uniformed individuals whose primary duties are the protection of facilities from acts that could endanger the health and safety of the public.
- b. All security alarms should annunciate in a continuously manned, onsite central alarm station and in at least one other continuously manned station not necessarily onsite . . ."

The criteria and requirements for procedural measures and design features for sabotage protection are based on certain assumed levels of potential threats. These are described in ANSI N18.17 as follows:

"Planning for industrial security should provide flexibility to respond to a wide variety of potential threats. In preparing this standard, specific consideration was given to potential threats posed by: (1) a single disgruntled employee who is authorized to have access to the plant and who is familiar with the details of construction and operation of the plant; (2) a single fanatic or mentally deranged person, either an authorized employee or an outsider, whose knowledge of the plant may range from none to intimate familiarity; (3) a small group of discordant individuals, some of whom may be armed, not normally authorized access to the plant, but who are intent on perpetrating acts of sabotage against the plant; and (4) spontaneous and undisciplined actions of a relatively large group of people involved in mob activities associated with acts of civil disturbance. Although it is clear that other potential threats may exist or develop, conscientious application of measures designed to protect against the threats discussed above will provide substantial protection against other postulated threats.

"Protection against actions associated with deliberate assaults by trained para-military groups, or military units of a foreign power is not covered by this standard. Protection against such actions is the responsibility of the U.S. Government."

With respect to the requirement for on-site security personnel, ANSI N18.17 recognizes that:

"Unique site-related characteristics associated with a given facility and the nature and extent of off-site security forces, as well as the reliability and availability of timely response of such forces, may, in unusual cases, justify the acceptability of relying on off-site security forces as substitutes for the plant security force required above. Exceptions to the requirement for plant security forces shall be justified for each facility for which an exception is proposed. In no case shall the response requirements of Sections 3.3.4 and 3.4.4 be compromised by any exceptions proposed under this paragraph.

"As noted in Section 1 of this standard, the combination of administrative controls, physical protection devices and security forces to be applied in the security program for each facility should be tailored to the unique characteristics of the site and the specific needs of the owner organization, including consideration of the nature of any threat associated with the plant site and its environs.

"The use of augmented physical protection devices, electronic surveillance, tightened administrative controls and close liaison with off-site security forces may serve to reduce the size of plant security forces that would otherwise be required."

IV. LICENSING ACTIVITIES AND SCHEDULE

A schedule of licensing activities for this program is outlined in Figure 9. The licensing concept is based on the utilization of the NRC procedures that provide for separation of plant and site licensing (CFR 50, Appendix M). These procedures are being applied in licensing actions for the manufacture of Floating Nuclear Plants by Offshore Power Systems. They provide for factory production of a specific number of standardized nuclear power plants under a single manufacturing license, based on postulated parameters of the eventual operational sites.

Since the proposed subsea nuclear power systems are essentially complete, "packaged" plants when they leave the final manufacturing facility, and since plant installation on the sea floor entails only minimal, if any, site preparation or on-site construction, it is suggested that the "construction" application-review and the "operating" application-review can be combined in a single proceeding, as indicated in Figure 9.

Separate construction-operating license proceedings will be required for each subsea plant site. These proceedings will focus on the specific site and site-related matters, including the extent to which that site falls within the postulated parameters specified at the manufacturing license stage.

The manufacturer (Atomics International) will apply for a license to fabricate a specified number of subsea nuclear power plants, of standard design, for installation and operation at a number of offshore subsea sites, to be specified later. An envelope of postulated operating site parameters, representative of sites located from 20 to 200 miles offshore and in water depths from 600 to 3000 ft, will be specified in this application.

A pre-application review of the subsea plant concept and the manufacturing plan will be conducted with NRC and other appropriate government agencies prior to the filing date, in order to assure that the application completely addresses all areas of special interest or concern.

The application for manufacturing license will meet all the requirements of 10 CFR 50.34 (a) (1) through (9) and 10 CFR 50.34a (a) and (b), except that the preliminary safety analysis report will be designated a "design report," and any

Figure 9. Subsea Nuclear Power System Licensing Schedule

required information or analyses relating to site matters will be predicated on the postulated site parameters.

Environmental reports will be submitted with the application, as required of applicants for construction permits under CFR 51, except that such reports will be directed at the manufacture of the reactors at the manufacturing sites, and, in general terms, at the installation and operation of the reactors at hypothetical seafloor sites having characteristics that fall within the postulated site parameters. The related draft and final detailed statements of environmental considerations prepared by the NRC will be similarly directed.

CFR 50, Appendix M, requires that the application for manufacturing license include a description of the proposed design of, and the site parameters postulated for, the reactors, including, but not limited to, the principal architectural and engineering criteria for the design, and the identification of the major features or components incorporated therein for the protection of the health and safety of the public. It requires information pertaining to design features of the proposed reactors that affect plans for coping with emergencies in the operation of the reactors. It further requires that any safety features or components which require research and development be described, and that research and development programs, reasonably designed to resolve any safety questions, be identified.

Further technical or design information, as may be required to complete the design report and which can reasonably be left for later consideration, will be supplied in a supplement to the design report. The final design of the subsea nuclear power systems will be submitted to the Commission as soon as it is completed, in the form of an application for amendment of the manufacturing license.

The owner-operator of the reactor plant will file application for the construction-operating license, submitting the necessary information relating to the specific operating site, and to plant transport, installation, recovery, and refueling activities.

This application need not contain information or analyses that have previously been submitted to the Commission in connection with the application for a manufacturing license. It will contain, in addition to the information otherwise

Figure 10. Nuclear Power Plant Licensing and Construction Process
(From U.S. Nuclear Regulatory Commission Annual Report 1975)

required by 10 CFR 50.34 (a) and 10 CFR 50.34a, sufficient information to demonstrate that the site on which the plant is to be operated falls within the postulated site parameters specified in the manufacturing license application.

Figure 9 shows the schedule of NRC review actions, and the related plant engineering and manufacturing activities. The scheduled times for NRC review are taken from Light Water Reactor licensing experience, ⁽¹⁴⁾ as presented in Figure 10. Since the installation of the proposed subsea plants does not entail significant site preparation or on-site construction activities, a substantially shorter overall licensing schedule can be achieved. The times required for regulatory reviews and approvals are expected to be schedule controlling.

The principal licensing application documents required at each stage of the proceedings are the Environmental Reports and the Safety Analysis Reports. Appendices A and B present outlines of the contents of these reports for the activities involved in this program. These outlines are based on Regulatory Guides 4.2⁽¹⁵⁾ and 1.7,⁽¹⁶⁾ published by the NRC as guidelines for the preparation of similar reports for Light Water Reactor projects.

For the Manufacturing License application, a Preliminary Safety Analysis Report (Design Report) and an Environmental Report, directed primarily at the manufacturing activities, manufacturing sites, and related environmental impacts, are required [CFR 50.34 (a)].

For the Construction-Operating License application, a Final Safety Analysis Report and an Environmental Report, directed at the transport, installation, operation, and refueling activities, and related to specific operating sites, are required [CFR 50.34 (b)].

Additional documents that will be required during the licensing process include a Manufacturing Plan, a Quality Assurance Program, a Special Nuclear Materials Management Program, and Security Plans for the protection of special nuclear materials and for protection of the reactor plant against industrial sabotage. These documents are important to the licensing process, but are not expected to be schedule controlling.

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APPENDIX A
ENVIRONMENTAL REPORT OUTLINE
SUBSEA NUCLEAR POWER SYSTEM

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 - 1.1 System Demand and Reliability
 - 1.2 Consequences of Delay
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 - 3.1.1 Reactor
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- 5.0 Environmental Effects of Base-Operational Station Operation
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- 6.0 Effluent and Environmental Measurement Program -- Base-Operational Sites
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 - 8.1 Benefits
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- 9.0 Alternative Energy Sources and Sites
 - 9.1 Alternatives Not Requiring the Utilization of Subsea Nuclear Power System
 - 9.2 Alternative Subsea Nuclear Power Systems
 - 9.3 Cost-Effectiveness of Candidate Site-Plant Alternatives
 - 9.4 Costs of Alternative Power Generation Methods
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- 11.0 Summary Cost-Benefit Analysis
- 12.0 Environmental Approvals and Consultation
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PRELIMINARY SAFETY ANALYSIS REPORT OUTLINE
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- 5.0 Reactor Coolant System and Connected Systems
 - 5.1 Summary Description
 - 5.2 Integrity of Reactor Coolant Boundary
 - 5.3 Reactor Vessel
 - 5.4 Component and Subsystem Design
- 6.0 Engineered Safety Features
 - 6.1 Engineered Safety Feature Materials
 - 6.2 Reactor Modules System
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- 7.0 Instrumentation and Control
 - 7.1 Introduction
 - 7.2 Reactor Trip System
 - 7.3 Engineered Safety Feature Systems
 - 7.4 Systems Required for Safe Shutdown
 - 7.5 Safety-Related Display Instrumentation
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- 8.0 Not Applicable
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- 11.0 Radioactive Waste Management — Base Station
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 - 11.4 Radiological Monitoring and Sampling System (Including Operational Site)
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 - 13.8 Industrial Security at Base Station
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- 14.2 Operational Site
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