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DESIGN REPORT

3000-kwe

SUBSEA NUCLEAR POWER PLANT

AEC Research and Development Report

*Prepared for the U.S. Atomic Energy Commission,
Division of Space Nuclear Systems,
under Contract Number AT(04-3)-701*

MASTER

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ABSTRACT

This design report describes a subsea nuclear power plant for use in offshore oil and gas fields, capable of generating 3000 kwe of electrical power, unattended and continuously, for 4 years. The nuclear power plant consists of three major items: (1) a reactor module, containing the zirconium hydride reactor and primary NaK coolant loop, (2) a generator module, containing the electrical generating equipment and the organic Rankine power conversion system, and (3) a waste heat exchanger, to dissipate the waste heat to the surrounding seawater. A fourth necessary item, although not a part of the nuclear power plant, is the transportation-emplacement barge, used to transport the nuclear plant to its location and to provide an emplacement vehicle on the ocean floor.

The safety aspects of the proposed subsea nuclear plant are compared with commercial light water power reactors. Due to the fact that the subsea nuclear power plant is hermetically sealed, its safety and reliability margins exceed typical commercial plants. The endothermic behavior of the zirconium hydride fuel also precludes any possibility of a fuel meltdown incident.

The report also discusses the current development status of each major component, and compares the subsea component design criteria with previously tested hardware.

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I. INTRODUCTION

This technical report presents the results of a study on the design of a 3000-kwe subsea nuclear power system for offshore oilfield application. As part of the Nation's goal to achieve energy self-sufficiency in the 1980's, substantial increases must be achieved in the production of oil and gas from offshore regions. The federal government has announced plans to accelerate the offshore leasing program. Future leases will include areas in deeper waters, areas involving environmental problems, and regions subject to severe operating conditions (e. g., Arctic icing).

These factors, as well as other production advantages, have caused virtually every major producer to take a hard look at the current state of development of subsea production systems, and the probable need to make more widespread use of subsea technology in the not too distant future. The pacing constraint is primarily an economic one. When the need is more acute and the probability of payout is favorable, subsea production will accelerate. Currently, about a hundred wellheads have been completed on the ocean floor. In the past year, the increase has been $\sim 30\%$; this sizable growth indicates that subsea installations are becoming commercially attractive in an increasing number of circumstances.

These wells now produce to nearby production facilities, on shore or on fixed platforms; and they have only nominal requirements for pumping production fluids, and no provision for gas compression. Conventional diesel-powered generators can satisfy current needs for subsea electric power. New developments in underwater equipment and techniques, now in the works by the major oil and ocean engineering firms, will soon provide the capability to install manifolds, pumps, compressors, separators, and other production equipment on the ocean floor. The single major item not being actively developed is a reliable, self-contained, economic subsea power supply of suitable capacity. This report describes a nuclear subsea power system designed to fill this gap in subsea production technology.

A recent applications study selected 3000 kwe (4000 hp) as a typical power requirement for future subsea production systems. That would suffice to

boost 50,000 barrels oil per day (BOPD) at 1000 gas/oil ratio (GOR) through a 25-mile long, 16-in. pipe, rising 1500 ft; or, it would pump the same flow up a 3000-ft deep marine riser; or, it could provide downhole lifting power to a field of 20 wells. The choice of 3000 kwe seems realistic and representative for future use; and the vessel weights and sizes seem reasonable, in view of the installation procedures to be used.

The subsea nuclear power concept described is a spinoff of space power technology developed at AI for the USAEC, over the past 17 years. This technical approach appears best suited to meet the requirements of long-term, highly reliable, unattended undersea operation, since these requirements have been prime objectives in space nuclear power development throughout the program, and have, in large measure, been achieved in small space power systems.

The zirconium hydride reactor was developed for highly reliable, unattended operations in space, and is readily adaptable to the undersea application. The organic Rankine power conversion system (PCS), which can be hermetically sealed, has also demonstrated long life, unmaintained operations, in smaller systems. This concept joins these two proven systems, and provides a reliable, economical, subsea power system to produce 3000 kwe for 4 years of full-power operation between refueling and refurbishment periods.

II. POWER SYSTEM DESIGN AND PERFORMANCE

This section presents the overall design and operating characteristics of the nuclear power system.

A. POWER SYSTEM CONFIGURATION

The subsea nuclear power system is designed to provide 3000 kw of electrical power (4160 v, 400 Hz) at the switchboard for a minimum unattended operating life of 4 years. The overall system, Figure 1, is composed of three major systems, designed for mounting upon a barge which can be placed on the sea floor. The three major systems (i. e., the reactor and primary coolant system, the organic Rankine power conversion system, and the waste heat rejection system) are discussed in detail in the following sections.

The principal system design criteria are safety, reliability, and minimum cost. Current technology and/or proven design approaches are used throughout.

Figure 2 is a simplified schematic of the nuclear power system, and Table 1 summarizes the overall system performance. Thermal energy is generated in the zirconium hydride reactor, and is transferred to the organic boiler via a liquid metal coolant loop. Here, the liquid metal coolant, NaK, a eutectic mixture of 78% potassium and 22% sodium, 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 400 Hz generator, producing 3300 kwe gross. The toluene vapor from the turbine exhausts to a regenerator, then to a condenser. Waste heat from the power system is then dissipated to the sea water through the 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 (-5°F). The waste heat rejection system is totally contained, and rejects heat to the sea water by natural convection around the heat exchanger tubes. The entire system is designed to withstand the pressure of the external sea water head, both externally and internally, so that a leak in the seawater side of the waste heat rejection system will not affect the condensing portion of the system.

Figure 1. 3000-kwe Subsea Nuclear Power Plant

Figure 2. Nuclear Power System Schematic

TABLE 1
PERFORMANCE SUMMARY

Electrical Power (kwe)	
Gross	3300
Net	3000
Thermal Power (Mwt)	
Reactor	10.0
Primary Heat Losses	0.2
Boiler	9.8
Temperatures [$^{\circ}\text{C}$ ($^{\circ}\text{F}$)]	
Reactor Outlet	427(800)
Reactor Inlet	316(600)
Turbine Inlet	371(700)
Condenser Outlet, Power Conversion System	42(107)
Condenser Inlet, Heat Rejection System	21(70)
Sea Water	10(50)
System Pressures [kg/cm^2 (psia)]	
Primary NaK	1.4(20)
Secondary Organic	42.9(610)
Waste Heat Rejection Organic — dependent upon design depth	
System Flow Rates [kg/sec (lb/sec)]	
Primary NaK	102.5(226)
Secondary Organic	15.4(34.0)
Heat Rejection Organic	183.3(404)
Efficiencies (%)	
Overall System	30
Power Conversion System	33
System Lifetime (years)	4

TABLE 2
WEIGHT SUMMARY
(lb)

Reactor Module Weight	
Pressure Vessel (1200-ft depth capability)	119,560
Shield	112,160
Core	7,740
Drive Mechanisms	960
NaK Piping System	13,100
Core Vessel	9,320
Heat Exchanger (Filled at 100°F)	4,050
Structure, Miscellaneous	<u>2,800</u>
Total	269,690
Generator Module Weight	
Pressure Vessel (1200-ft depth capability)	126,040
Turbine-Alternator-Pump	6,310
Regenerator and Condenser	13,710
Piping Systems, Organic, Insulation	3,340
Batteries, Wiring, etc	2,050
Structure, Miscellaneous	<u>3,100</u>
Total	154,550
Seawater Heat Exchanger	
Piping, Coolant	11,440
Structure, Miscellaneous	<u>2,000</u>
Total	13,440
System Total	437,680

All three major systems are hermetically sealed. The primary NaK system uses an all-welded stainless steel construction. Both organic loops are also all-welded systems. Table 2 summarizes the weight of each major system.

Figure 3 shows the organic Rankine cycle operating conditions for the 3000-kwe net subsea power station. The reactor generates 10 Mwt, and transfers 9.8 Mwt to the organic loop. Here, through control of the turbine bypass valve and nozzle flow area, sufficient organic vapor is allowed to enter the single-stage axial-flow turbine to maintain the generator frequency within preset limits. The turbine vapor exhausts to the regenerator, which has a design effectiveness of 95%. From there, the vapor is condensed in the condenser, which contains a hot well for storage of the excess organic needed during startup, and to permit volumetric expansion as the fluid is heated. A submersible organic pump is located in the hot well. This pump provides the NPSH for the main organic pump, which is mounted on the same shaft as the turbine-generator assembly. The main organic pump then pumps the organic through the regenerator to the boiler, and then to the turbine. Waste heat is removed by another organic loop which pumps the waste heat fluid to a large heat exchanger located on the deck of the barge. Figure 4 shows the temperature-entropy diagram for this supercritical organic cycle.

B. REACTOR MODULE

The reactor module (Figure 5) consists of the reactor, NaK piping, NaK pump, boiler, reactor control, radiation shield, and pressure vessel. The reactor is made up of 19 fuel assemblies (Figure 6). Each fuel assembly contains 72 fuel elements in a hexagonal pattern, with a central port for control rod access. The volume between the edge of the 19 fuel assemblies and the core vessel contains beryllium reflector pieces for neutron economy and radial power flattening. In addition, a 3-1/2 in. thick cylinder of borated stainless steel surrounds the core to provide thermal shielding, so that the temperature of the serpentine concrete shield remains below 800°F (427°C). The fuel assemblies are contained within a stainless steel reactor vessel which has a 47-3/4 in. (1.2 m) outside diameter and is ~9 ft (2.75 m) long.

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Figure 3. 3000-kwe Organic Rankine Cycle

Figure 4. Organic Rankine Cycle Temperature-Entropy Diagram

Each fuel element contains seven fuel slugs, 4 in. (10.16 cm) long, containing 10.5 wt % of $\sim 63\%$ enriched U^{235} , with a 1-1/2 in. slug of beryllium located at each end for neutron economy and axial power flattening. The fuel and beryllium end pieces are clad in 0.025-in. (0.635-cm) thick stainless steel tubing, with NaK bonding to improve the heat transfer characteristics. Table 3 gives the key nuclear parameters.

The reactor generates 10 Mw of thermal power which is removed by the primary loop NaK. The NaK enters the bottom of the reactor at 600°F (316°C) and exits at the top at 800°F (427°C). The reactor is oriented vertically to provide natural convective flow in the unlikely event of loss of pump power (see Section III). The primary loop piping system is stainless steel.

The boiler is a stainless steel, tube-and-shell design, with the NaK on the shell side and the higher-pressure organic fluid (toluene) in the tubes. The boiler has a semi-toroidal shape, ~ 14.6 ft (4.44 m) long and 12 in. (30.5 cm) in diameter, and is located in a horizontal plane, a few feet above the reactor.

The NaK is pumped by an ac helical induction pump (Figure 7) which has a flow rate of 2070 gpm (7835 l/min) at a head of 10 psi (0.703 kg/cm²). This type pump was selected because it has no moving parts or seals, and should therefore provide high reliability for long-term operation. The NaK loop pressure losses are 6 psi (0.42 kg/cm²) in the reactor, 2 psi (0.14 kg/cm²) in the boiler, and 2 psi (0.14 kg/cm²) in the NaK lines. The pump is 60 in. (152.4 cm) long and 24 in. (61.0 cm) in diameter. It has an efficiency of 19.5%, requiring an input of 47.2 kwe. During startup, the pump is powered by batteries through an inverter.

Each of the 19 fuel assemblies is provided with a boron stainless steel control rod. The control rods are ganged, three to a drive system, except for the center rod, which has its own drive system. The drive system utilizes a rack and pinion type mechanism, driven by a small ($\sim 1/10$ hp) ac electric motor through a magnetic clutch. All components in the drive systems are standard off-the-shelf items, all of which have a successful history of operation in similar environments. The control rods are attached to the drive systems through nested formed bellows which provide the NaK seal.

Figure 5. Reactor Module

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Figure 6. Fuel Assembly

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TABLE 3
NUCLEAR PARAMETERS

U^{235} (kg)	69.3
Enrichment (% U^{235})	63
H/Zr (atom ratio)	1.66
Cold, Clean Excess (\$)	26.70
Shutdown Margin (\$)	5.80
Control Worth (\$)	32.50
Peak Fuel Burnup (met. at. %)	1.30
Average Fuel Burnup (met. at. %)	0.71

Figure 7. ac Helical Induction
NaK Pump

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Figure 8. Generator Module

The reactor is surrounded on all sides by a serpentine concrete radiation shield. After a short time (~ 100 hr) bake-out, serpentine concrete retains its remaining water of hydration at an almost constant level. Its effectiveness for neutron shielding is therefore stable at the design temperatures of this application. The functions of this shield are to: (1) minimize activation of the containment vessel, (2) prevent activation of the surrounding seawater in excess of MPC, (3) allow maintenance in the vicinity of the reactor shortly after it is shut down, (4) prevent radiolytic deterioration of the organic working fluid, and (5) permit access to the generator module during reactor operations.

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 same end is provided for organic piping and the necessary control and instrumentation leads.

C. GENERATOR MODULE

The organic Rankine power conversion or generator module consists of the turbine, regenerator, condenser, hotwell, organic pumps, generator, and pressure vessel. Figure 8 shows a typical generator module arrangement.

The organic working fluid (toluene) is heated in the boiler to 700°F (371°C) and 600 psia (42.2 kg/cm^2), and is transported into the generator module. Here, the turbine bypass valve, in conjunction with the turbine admission throttle, controls the flow of vapor to the turbine to maintain the generator frequency at 400 Hz. The vapor expands in the turbine to a pressure of 1.75 psia (0.123 kg/cm^2) and a temperature of 466°F (241°C). The turbine delivers shaft power to the generator to produce ~ 3300 kwe gross. The turbine is a single-stage, supersonic, axial impulse, full-admission design, rotating at ~ 8000 rpm. It has a diameter of 34 in. (0.86 m), and a blade height of 5.2 in. (13.2 cm). The turbine efficiency is $\sim 78.3\%$. A 400 Hz system is used to minimize equipment size and weight. It also eliminates the need for a gear box and associated seals, since the generator can be driven directly from the turbine, and thereby permits a totally hermetically sealed organic system.

The vapor leaving the turbine passes through the regenerator, where it preheats the liquid toluene before the latter enters the boiler. The vapor leaves the regenerator at a pressure of 1.50 psia (0.11 kg/cm^2) and a temperature of 128°F (53°C). Because of the low vapor density and the desire to reduce pressure losses, the regenerator has a large frontal area and a short cooling path length. The finned tube and shell regenerator measures $\sim 2\text{-}1/2$ by $6\text{-}1/2$ by $6\text{-}1/2$ ft (0.76 by 2.0 by 2.0 m).

The vapor is cooled and liquified in the condenser at a temperature of 107°F (42°C), corresponding to a vapor pressure of 1.25 psia (0.09 kg/cm^2). The heat sink for the condenser is an organic fluid also (e. g., a mixture of 50% ethylene glycol and 50% water) that is pumped through at a rate of 404 lb/sec (183 kg/sec). The condenser is a tube-and-shell type, with the high-pressure organic fluid in the tubes. The dimensions of the condenser are 3 by $4\text{-}1/2$ by $4\text{-}1/2$ ft (0.91 by 1.37 by 1.37 m).

The liquid toluene is heated to 375°F (191°C) in the regenerator, prior to entering the boiler. This preheating greatly increases the cycle efficiency. The liquid is further heated in the boiler from 375°F (191°C) to 700°F (371°C).

The generator supplies ~ 260 kwe auxiliary power to the NaK pump, organic pumps, and control and instrumentation systems, as well as 3000 kwe to the external load.

The pressure vessel for the organic Rankine system generator module is also made of ASTM A516; its thickness and frame spacing depend upon its design operating depth. The module is flanged at one end to facilitate removal of large items of equipment for refurbishment, repair, or replacement. A smaller flanged opening is provided for the organic piping, and instrumentation and control leads from the reactor module. The waste heat exchanger piping also penetrates the module side, and the electrical power leads leave the module through a fixed electrical connection.

As the design and placement of the generator module provides for limited manned entry during reactor operations, a service capsule mating hatch is provided on the top of the pressure vessel. The vessel is designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for nuclear

containment, and also in accordance with the American Bureau of Shipping codes for manned subsea habitats.

D. WASTE HEAT EXCHANGER SYSTEM

The waste heat exchanger system is a totally closed system, using a low-freezing-point organic mixture (e. g., 50% ethylene glycol and 50% water), which removes ~ 7.0 Mw of thermal energy from the condenser and dissipates the energy to the sea water via a tube bundle located on the deck of the barge between the reactor module and the generator module. The tubes have headers to minimize the system pressure drop by providing numerous parallel flow paths. The temperature rise in the waste heat loop is minimized to achieve maximum overall cycle efficiency. Heat rejection is by natural convection of the sea water around the tubes. The tubes are designed to withstand a pressure differential, either internally or externally, equivalent to the hydrostatic head of the sea water at design depth. The system will normally be operated with a slight Δp across the tube, with the external sea water pressure being slightly higher than the internal organic pressure.

The organic mixture enters the condenser at 70°F (21°C), and exits at 90°F (32°C). It is then cooled in the tube bundle by the 50°F (10°C) sea water. The low ΔT requires a large flow rate, ~ 400 lb/sec (183 kg/sec), to reject the 7.0 Mw of thermal energy. The organic pump supplies the 20 psi (1.4 kg/cm²) pressure rise necessary to maintain the flow rate.

E. CONTROL AND OPERATIONS

The nuclear power system is designed for remote startup, shutdown, and restart. It is also designed to be load-following, although a constant load is expected to be maintained through the use of recirculation in the oil pumps and gas compressors.

Once firmly implanted on the sea bottom, and after all power and instrumentation connections have been made and verified, startup of the nuclear system can begin. Startup can be accomplished remotely from the supervisory station, the service vessel, or manually from the operating panel in the generator module.

1. Startup

The startup of the power plant involves bringing the reactor up to low power, preheating the components of the system to their operational temperatures, and then bringing the entire plant to rated operating power. The time for startup is limited by the time required to bring the secondary heat transfer loop to operating temperature, using heat produced by the reactor. This should be of the order of 1 hr at 10% power. The reactor power is brought up to some nominal level (10 to 20% full power), and held constant while the secondary loop is heating up. The flow of the NaK must be sufficient to assure adequate heat removal capabilities from the core without having a large temperature rise through the core.

The reactor is brought to power by inserting reactivity through programmed control rod withdrawal. Flux monitors are used to measure the reactor period, which is used in the safety instrumentation to limit the reactivity rate to a preset value. The simplest startup mode is achieved by using a constant rate of reactivity insertion. Should it become desirable to use multiple input reactivity rates to achieve faster startup, the flux monitors can be used to determine the appropriate reactivity rate switching points.

As the reactor starts producing sensible heat, the turbine bypass valve and the main organic pump bypass valve are opened, and the booster pump is started, using auxiliary battery power. Heat from the reactor is transferred to the NaK; and, as the NaK temperature rises, there will be some heat transfer to the toluene. The rate of increase of temperature of the toluene will be slow at first, since much of the heat transferred to the toluene will be used to heat up the equipment in the secondary loop; so that this equipment, except for the turbine, will be brought up to temperature along with the toluene. When the temperature and pressure of the toluene have been increased so that vaporization will occur in the boiler, then the turbine throttle valve is opened to admit a small flow of vapor through the turbine, and this brings the turbine up to temperature. As the turbine temperature increases, the flow of toluene through the turbine is increased, and the flow through the bypass valve is decreased. The main toluene pump is mounted on the turbine shaft, and this will come on line as soon as the turbine turns. The pressure at the turbine

inlet is controlled by the turbine bypass, and this is used to accommodate the increased head when the main pump comes on line.

After the turbine is brought up to speed and synchronized, power is supplied to the plant auxiliaries by the generator. Then the reactor is brought up to full power, and external loads are switched to the generator.

2. Power Operations

Power is supplied by the nuclear system for pumping oil and/or compressing natural gas, plus operating other electrical equipment in the oil-gas production installation. Power is also supplied to the auxiliary and instrumentation and control circuits necessary to operate the power plant. The only equipment sensitive to changes in the input voltage and frequency are the instrumentation and control systems (both the nuclear plant and the production plant). These will operate from a battery-supplied, ac-regulated circuit. The batteries will be connected to the generator for continuous charge. Therefore, it is not necessary to maintain tight constraints on the generator output. This allows considerable flexibility in the design of the power plant control and instrumentation systems.

The voltage at the output of the generator is maintained constant by controlling the exciter voltage. The frequency of the output is determined by the speed of the turbine shaft, and a mismatch between the load and the power supplied is reflected in changes of the shaft speed. The accuracy required of the control system is governed by the accuracy required for the shaft speed and the generator frequency. The toluene pump is shaft mounted, and the effects of shaft speed on the pressure and flow rate of the toluene must be considered.

The parameters of principal interest for the proper operation of the power plant are: (1) the reactor outlet temperature, (2) the temperature rise through the reactor core, (3) the temperature and pressure at the inlet of the turbine, and the maximum temperature of the toluene.

The turbine is supplied with a throttle valve to regulate the inlet conditions to meet the immediate output requirements. This, together with the heat capacity of the working fluid, supplies the immediate load-following requirements. The pressure and the inlet to the throttle is maintained by opening or

TABLE 4
AUTOMATIC SHUTDOWN SIGNALS

High Reactor Flux

High Reactor Outlet Temperature

Low Reactor Coolant Flow

Loss of Electrical Power

Water in the Modules

Reactor Period (during startup only)

Low Organic Flow

Acceleration (earthquake)

closing the turbine bypass valve. Floating proportional control is used to match the reactor power with the load requirements, according to the appropriate partial-load characteristics. A deadband is supplied in the power feedback loop to minimize the necessity of actuating the control rod drive to follow small changes in the load. Small changes are accommodated by changes in the cycle performance due to the throttle and turbine bypass valves actions, and by the negative temperature coefficient of the reactor.

Changes in the load requirements are due to variations in the power requirements for the oil pumps and the possible loss of one or more pumps on line. When the load changes are large, it is necessary to feed back the power requirements to vary the reactor power level. In addition to power matching, variation in the toluene temperature at the boiler outlet is used as a trim control to assure proper operating conditions in the working fluid.

Changing the power in the primary loop requires changing the flux level in the reactor by moving the control rods, and also changing the NaK flow rate to maintain the proper temperature rise through the reactor core. The power feedback from the partial load schedule and the boiler outlet temperature trim signals are used to vary the NaK flow rate by controlling the NaK pump. The flux controller also includes a trim signal from the reactor outlet temperature, so that an increase in the reactor outlet temperature above preset limits causes a decrease in the flux level.

The temperature of the toluene is closely coupled with, and limited by, the outlet temperature of the NaK. Therefore, the feedback of both the maximum NaK temperature and the maximum toluene temperature for the control of the reactor power serve to limit the toluene temperature. This, in turn, limits the degree of pyrolytic breakdown of the toluene.

3. Shutdown

The nuclear system is designed to be shut down via remote control from the supervisory station or from the service vessel. Manual shutdown capability is also provided for at the control panel in the generator module. An automatic shutdown (scram) system is also provided. This uses 2-out-of-3 coincidence to minimize the possibility of inadvertant scrams. Table 4 lists the parameters which can cause automatic shutdown.

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Figure 9. Dose Rate at Full Power vs Sea Water Distance From Reactor
Module Pressure Vessel Surface

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If the nuclear system is shut down, either manually or automatically, and the reason for the shutdown has been cleared, the system may be restarted by the same procedures used for the initial startup.

4. Radiation-Activation

During full-power operations, any assigned workman can enter the generator module for a cumulative period of 48 hr during any 3-month period without exceeding the radiation dose level for nuclear workers established by the USAEC. By rotating workmen, a full work schedule can be maintained, if this should be necessary or desirable. The maximum dose in the module is <15 mr/hr at the reactor control panel and main switchgear.

The radiation level at the surface of the reactor module pressure vessel during full-power operations will be <800 rem/hr. Figure 9 shows how this dose rate decreases with sea water distance from the surface of the pressure vessel. The neutron flux at the vessel surface is low enough to limit sea water activation to less than the MPC allowed (10 CFR 20), assuming a minimal flow of sea water of 1 cm/sec (0.02 kn). It is also low enough to prevent excessive neutron activation of the pressure vessel, so that refurbishment work may be conducted without undue radiation hazards. The neutron flux at the liquid metal and generator components location is $\sim 2.4 \times 10^6$ n/cm²-sec, which will not cause excessive activation. Furthermore, the neutron flux and gamma ray doses are low enough that no significant radiolytic damage of the organic coolant will occur.

After the reactor has been operated, a significant fission product inventory will be accumulated in the core, and the primary loop NaK will be activated. The fission product dose rate can be maintained at acceptable low levels, on and around the barge, by keeping the lower half of the reactor module under at least 5 ft of water; however, the dose rate, on and around the barge, when surfaced, from the NaK in the upper half of the reactor module does not permit uncontrolled access until several days after shutdown. Earlier access to areas on and around the barge can be provided by installing a temporary shield around the upper portion of the reactor module. About 10 in. (25.4 cm) of concrete will reduce the dose rate an order of magnitude; this is equivalent to ~ 2 days wait time. Figure 10 shows the dose rate from the NaK in the reactor module at various distances from the reactor module, without water or other shielding.

Figure 10. Dose Rate Due to Na^{24} vs Time After Shutdown for Various Distances From Reactor Pressure Vessel (in air)

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F. INSTALLATION

The preconceptual design utilizes a barge to both transport the nuclear power system to the production site and for placement on the ocean bottom. The barge is ~48 by 84 ft, and is designed to be towed to the site. Typical transportation design considerations include collision, tow line breakage, and overturning due to wind and/or wave action.

Prior to placement of the barge on the ocean bottom, the sea floor will be prepared to receive the barge. The barge is then towed to the site. At the site, the barge is attached to the sea floor mounting, and is placed on the mounting structure using a combination of controlled bouyancy and downhaul lines. Once securely in place on the sea floor mounting, additional negative bouyancy is added by flooding additional ballast tanks to provide a large negative bouyancy. Figure 11 illustrates the procedure for placement.

The nuclear system is now ready to be connected to the production station. This is accomplished through the use of subsea cables. One or more power cables, plus instrumentation cabling, will be connected to the power system. It is anticipated that the generator module will be occupied at this time, and that final connections will be made manually.

At this point, the nuclear system may be started, as described in Section II-E.

Figure 11. Installation of Subsea Nuclear Power Plant

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III. SAFETY AND RELIABILITY

This section of the report discusses the safety and reliability of the nuclear system. The reliability of the system is shown to be potentially very high, by comparing the system with demonstrated hardware.

A. NUCLEAR SYSTEM SAFETY

During pre-operation phases, the system is essentially nonradioactive, and can therefore be handled and transported with only minimal restrictions. The major safety concern is one of preventing inadvertent reactor criticality. Once the reactor is assembled in the reactor module, the only way to achieve inadvertent criticality is by accidental withdrawal of the control rods. This is precluded by a mechanical ratchet device, even in the unlikely event that the reactor module is completely tipped over.

Low-power operation of the nuclear power system begins with acceptance testing, prior to shipment to the installation site. The entire system is checked out functionally. The reactor is brought to criticality; but, because of the short operating time and the low power, no significant fission products are generated, nor is any material activated. The total fission product inventory is <10 mCi, which results in a dose rate less than natural background at 1 ft (0.3 m) from the pressure vessel.

When the reactor is operating on its sea bottom mounting, there is no significant activation of either the sea water or the reactor module pressure vessel. Figure 9 shows the dose rate as a function of distance from the pressure vessel. The barge also precludes the access of fish, except for the smallest varieties, to the close vicinity of the reactor pressure vessel.

The primary safety design and operating requirements for any nuclear plant relate to the prevention of release of radioactive materials to the general public. Figure 12 compares a Safety Assurance Diagram for the ZrH subsea nuclear power plant with a typical commercial light water reactor (LWR) power plant. Initially, fuel (fission product) containment is maintained by the fuel rod cladding material. Stringent quality control procedures are used in the fabrication of reactor fuel elements. In the case of the ZrH fuel, the fuel elements contain NaK; therefore, any minute leak in the cladding, resulting

from manufacturing defects, is immediately obvious by the presence of NaK oxide on the fuel element surface. In addition, each element is leak-checked after final welding closures are completed.

In the event of cladding failure during reactor operation, some of the gaseous fission products will be released to the coolant stream. In the case of the ZrH fuel, this amounts to <3% of the total gaseous fission product inventory; while, in the UO_2 fuel used in water reactors, about 95% is released. However, in the typical commercial LWR plants, a small fraction of the coolant and any noncondensibles, including fission gas products, are removed and processed through a radioactive waste system. In the case of the ZrH plant, an all-welded NaK coolant system is used, and any gases de-entrain in the surge tank and are held until refurbishment.

The key radioactive materials containment barrier in both the ZrH system and the LWR systems is the primary coolant system. In the ZrH plant, this system is a hermetically sealed system, having no seals, drain lines, or valves. Although a typical LWR primary coolant system is of welded construction, there are valves to the various radioactive waste systems to remove noncondensibles from the coolant and to maintain the specified water chemistry in the primary systems. Valves can leak and be inadvertently opened, which can release radioactivity to the radioactive waste systems. This cannot occur in the proposed ZrH plant.

Another key design feature of the ZrH plant is its low pressurization (~ 20 psia) of the primary (radioactive) system, compared with the high operating pressure (>1000 psia) in the LWR systems.

In the unlikely event of a primary coolant system rupture, the oxidation of the NaK is limited by controlling the oxygen content of the cover gas in the modules in the second level of containment (i. e., Number 2 in Figure 12); an oxygen content of <5% will preclude any possibility of rapid oxidation, fire, or explosion. The integrity of the modules (i. e., the pressure vessel) is therefore not affected by a primary system leak. Fuel meltdown in a loss-of-coolant accident is not thermodynamically possible in the ZrH reactor, due to the endothermic desorption reaction (40 kcal/mole) as the hydrogen is driven off by the decay heat. In contrast, water reactors rely upon active systems,

ZrH SUBSEA NUCLEAR POWER PLANT

TYPICAL COMMERCIAL WATER POWER PLANT

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Figure 12. Safety Assurance Diagram Comparison

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the Emergency Core Cooling System (ECCS), to prevent a fuel meltdown during a Loss of Coolant Accident (LOCA). Both passive and active systems (hydrogen-oxygen recombiners) are relied upon to prevent buildup of pressure in the containment sphere. Failure of these systems could result in a breach of tertiary containment; whereas no systems, active or passive, are needed in the ZrH reactor module to prevent pressure buildups of sufficient magnitude to stress the pressure vessel. There is insufficient stored energy in the coolant to cause any significant gas pressure buildup.

In the extremely unlikely event that there is a breach of third level of containment (Number 3, Figure 12), there will be some minimal release of radioactive material to the environment. In the case of the commercial LWR power plants, a fraction of the gaseous fission products will be released, but all solid fission products will probably be contained. The released activated primary coolant water will also add to the potential off-site dose rate. In the case of the ZrH reactor, rupture of the pressure vessel will permit entry of high-pressure water. This, in turn, will cause a collapse of the primary coolant system, with a subsequent water-NaK reaction. This reaction, or flooding of the reactor core, could result in core disassembly, with the release of a few percent of the gaseous fission products to the ocean. Water-soluble solid fission products will also enter the marine ecological chain.

Based on preliminary assessments, the rapid decay of the radioactivity in the coolant (NaK), and the dilution of longer-lived fission products by the marine dispersal and ecological chain, will result in no harmful effect to the general public.

In summary, a comparison of the ZrH subsea nuclear plant with a commercial water reactor plant shows that the same barriers are utilized to preclude release of radioactive materials to the general public. However, in the case of the ZrH plant, no active or passive systems are required to prevent fuel meltdown or preclude pressure buildup in the containment vessel.

Further, tests conducted during the Aerospace Nuclear Safety Program demonstrated that, even in extreme cases in which U-ZrH fuel disassembly occurred, only a few percent of the total fission product inventory was released as gaseous radioactive material; this is an additional inherent safety feature of the ZrH reactor.

The dissociation of hydrogen from U-ZrH fuel, and the rapid permeation of hydrogen through the fuel cladding, as the temperature is increased, limits the extent and consequences of an inadvertent runaway of the ZrH nuclear system, in the event that control rods are accidentally withdrawn. In this extremely unlikely incident (in which all the safety circuits must be assumed to be inoperative), the fuel temperature would rise, causing hydrogen to be driven off. Hydrogen will readily permeate the stainless steel cladding, once its temperature exceeds 1000°F (538°C). This loss of hydrogen not only limits the maximum fuel temperature, but also rapidly removes reactivity from the core, causing the reactor to go inherently subcritical.

Four areas of concern in the operation of LWR systems are currently receiving additional attention by the Directorate of Regulatory Operations in the USAEC.* These are:

- 1) Fuel Performance — concern is the premature leakage of fuel rods resulting in releases of fission products, thereby making it more difficult to achieve effluent releases "...as low as practicable."
- 2) Valve Performance — primary concern is the relatively large number of valve malfunctions in safety-related systems.
- 3) Piping Support System Adequacy — primary concern is that reactor system designers may not have adequately designed for the effect of forces in the auxiliary piping support systems.
- 4) Blowdown Incidents — concern involves the unintentional discharge of significant quantities of radioactive primary coolant into the primary containment during reactor operations.

It is interesting to note that the four items of immediate concern in the LWR systems are not applicable to the ZrH plant. The first and fourth items are not applicable, because the ZrH system is completely welded, with no provision for blowdown or effluent release during operations. No valves are used in the operating plant; hence, Item 2 is not applicable. Item 3 does not

*WASH-1250, "The Safety of Nuclear Power Reactors (Light Water-Cooled) and Related Facilities" (July 1973) pp 3-20

totally apply, because there are no auxiliary systems connected to the ZrH radioactive primary system. However, as there are relatively high-pressure organic lines (600 psia), one must design and secure these lines so that any pipe rupture will not allow the free ends to whip, and sever any protective circuit wiring or damage any safety device that would fault the safety system.

The generator module is located a sufficient distance from the reactor module that limited manned access to the generator module is possible during operations. Manual inspection or maintenance is possible in a standard air environment, or by using breathing apparatus in the nitrogen environment. As all fluid systems are hermetically sealed, no repair on them is contemplated. The electrical control and instrumentation system can be maintained as required. The system is designed, however, for 4 years of unattended operation without maintenance.

The remaining safety questions relate to the transportation of the nuclear system back to port, and then to the refurbishment center, if located away from the port area. The reactor core and the activated NaK coolant emit gamma radiation, which decays with time after the reactor is shut down. The design of the barge is such that the reactor core is always located below the water line, and hence is shielded. The NaK in the boiler and in the pump, however, is exposed to the weather deck when the barge is afloat. These portions of the system must therefore be left underwater for ~ 7 days, to provide the shielding required for uncontrolled access to the ocean surface in the vicinity of the barge; alternatively, temporary additional shielding or exclusion areas around the barge can be provided to allow complete surfacing in < 7 days after reactor shutdown.

During the ZrH Reactor Development Program, an extensive nuclear safety project was conducted to evaluate the nuclear hazards associated with the fabrication, transportation, vehicle system integration, and operation of zirconium hydride reactor systems, and to develop engineering safeguards to enhance the safe use of such systems. In the course of the program, from 1962 to 1973, a number of analytical and experimental programs were accomplished. The programs most relevant to the undersea reactor include:

TABLE 5
REACTOR EXPERIENCE

	Past Experience	Current Design
Total Reactors Fabricated	16	
Total Test Time (hr)	43,000	
Reactor Tests		
Number Power Tested	6	
Power Level (Mwt)	to 1.0	10
Operating Temperature [$^{\circ}\text{F}$ ($^{\circ}\text{C}$)]	to 1300 (704)	800 (427)
Operating Durations (hr)	to 10,000	35,000
Fuel Burnup (met. at. %)	0.2 peak/0.1 average	<1.3 peak/0.7 average
Linear Power Density [kwt/ft (kwt/m)]	5.9 peak/3.38 average (19.4/11.1)	5.8 peak/3.1 average (19.0/10.2)
Average Peak Fuel Temperature [$^{\circ}\text{F}$ ($^{\circ}\text{C}$)]	1420 (771)	965 (518)
In-Core Fuel Element Tests		
Operating Temperatures [$^{\circ}\text{F}$ ($^{\circ}\text{C}$)]	500-1300 (260-704)	800 (427)
Fuel Brunup (met. at. %)	2.2	<1.3 peak
Effective Reactor Power (Mwt)	20	10
Linear Power Density [kwt/ft (kwt/m)]	≤ 16 peak (52)	5.8 peak (19)

- 1) Critical Configurations — A project was conducted to determine the reactivities of ZrH reactor cores under conditions of ocean impact, land impact, handling-shipping accidents, or launch-site incidents which could result in water reflection and/or flooding of the reactor core. Techniques were developed and successfully used for all safety problems associated with the fabrication, testing, transportation, handling, and launch operations of ZrH reactor systems.
- 2) Reactor Transient Behavior — Analyses to determine the effects of the maximum reactivity accidents were performed, supported by a program of reactor transient tests. As part of the test program, a ZrH reactor was submerged in water and subjected to an extreme transient initiated by rapid insertion of a large excess reactivity. The power excursion was allowed to proceed until terminated by core disassembly. Good agreement was obtained between the results of this experiment and predictions of peak power, total energy release, inverse period, reactivity behavior, and rate of core expansion. A series of experiments was performed on fission product release for ZrH fuel, which indicated that the total release from unclad samples was orders of magnitude less than the maximum theoretical release postulated.

Based on the extensive safety experience of the SNAP Program, and on a preliminary evaluation of marine systems, it is not expected that operation of the ZrH reactor system under normal or accident conditions will result in a significant nuclear safety hazard to marine life or to the general population.

B. POWER SYSTEM RELIABILITY

The subsea power system design draws heavily upon the space power technology, where systems have been designed and developed for long-term unattended operations. It is this technology that can be successfully and reliably adapted to the inner space applications of underwater oil operations.

The system reliability can be assessed by considering the experience that has been accrued on components of similar design. Table 5 summarizes the reactor test experience, and compares this experience with the design parameters of the 3000-kwe subsea reactor. As can be seen from Table 5, only

TABLE 6
ORGANIC RANKINE PCS EXPERIENCE

	Past Experience	Current Design
Power Range (kwe)	0.2 to 100 kwe tested; studies to several Mw	3 Mw
Frequency (Hz)	to 1200	400
Maximum Organic Temperatures [$^{\circ}$ F($^{\circ}$ C)]	650 to 825 (343 to 441)	700 (371)
Fluids	MIPB, Dowtherm A, CP-25 (Toluene)	CP-25 (Toluene)
Turbine Speed (rpm)	12,600 to 25,200	8,000
Turbine Diameters [in. (m)]	5 to 18 (0.13 to 0.46)	34 in. (0.86)
Bearing Types	Tilt pad hydrodynamic, oil roller	Oil-lubricated Roller
Life (hr)	100 kwe Design Life—50,000; smaller units tested to 36,000	35,000

lifetime and power level exceed the experience already successfully demonstrated by either a reactor test or by fuel element irradiations. In fact, due to the reduced coolant operating temperature to the 800°F (427°C) range, there is substantial margin in the subsea design, relative to the technology developed for the high-temperature, 1300°F (704°C), operating conditions of space power systems.

The major component which has not been demonstrated in the space program is the control rod drive; however, this component is fabricated from standard off-the-shelf components which have successfully demonstrated reliable operations in similar environments. Because of the small size core, this component will be designed to have large operating and safety margins.

Table 6 summarizes organic Rankine PCS experience. As can be seen from the table, the operating temperature, lifetime, and bearing concept of the current design is consistent with past experience.

In summary, the broad technology base provided by the ZrH reactor programs in the past 17 years should result in minimal development required to assure the reliability and safety of the relatively low-temperature reactor needed for this application.

Work on organic Rankine power systems has been in progress for several years, under various commercial and government-sponsored programs. This work has included boiler, regenerator, turbine, condenser, and pump design, analysis, and test. In addition, special emphasis has been successfully directed toward the development of various bearing types, and toward evaluation of a number of organic fluids.

IV. SUMMARY

This report describes the design and performance of a subsea nuclear power system for application in offshore oil and gas production. The system can be designed and built with minimum additional development, based on space nuclear power technology, components, and systems completed during the past 17 years. The concept is particularly well suited to meet the requirements of the offshore oil application for adequate safety, high reliability, and acceptable costs.

The system is designed to provide 3000 kwe, at 4160 v and 400 Hz, for a period of 4 years, in an automatic, unattended mode. At the end of this period, the system is returned to a refurbishment center, where the reactor core is replaced and other parts of the system are inspected, refurbished, or replaced, as required. With this periodic refueling and maintenance, the total system life should exceed 20 years.

The nuclear power system is largely contained within two separate pressure vessels, connected by an enclosed pipe chase. One vessel contains the reactor and liquid metal primary coolant system, while the other contains the organic Rankine power conversion system. These are designed to be mounted on a barge which provides both a means to transport the system and a mounting structure for the sea floor installation. A waste heat exchanger is mounted on the deck of the barge.

The nuclear power system is designed for unattended operations, and for automatic startup and shutdown. It can be operated automatically in a load-following mode, or controlled remotely with the use of supervisory and instrumentation circuits. The report also describes the safety features of the system, and makes a safety comparison with commercial, light water, nuclear power plants currently in use in the utility industry. This comparison shows that the subsea system has important safety features not found in commercial light water reactor plants.

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