THE NUCLEAR REACTOR

IMPACT ON SUBMARINE DESIGN AND OPERATION

BY

R. STRONG, M.A., AND R. NASH
(*Rolls-Royce and Associates, Ltd.*)

Introduction

The use of nuclear power for submarine propulsion, a prospect recognized by Enrico Fermi in the U.S.A. in 1939, reached its first practical demonstration with the commissioning of the U.S.S. *Nautilus* in 1955. Since then, the construction and operation of some hundreds of naval submarines, incorporating various developments of the basic system, have testified to the major performance benefits of nuclear power. Most significant is, of course, the underwater endurance which is increased from a few days to some months. Maximum speed, particularly when submerged, has been considerably increased. With so much time now spent submerged, the hull shape is naturally optimized for underwater performance—such shapes are now increasingly adopted for diesel-electric submarines. A number of other developments such as those in navigation systems and in air purification and regeneration systems have been important in realizing the full potential of the nuclear submarine. This paper however is concerned only with those effects on the submarine's design, operation and maintenance which are directly attributable to the nuclear reactor. Comparisons with the conventionally-powered submarine are mostly implied rather than stated.

The pressurized water reactor (PWR) has been employed in the vast majority of nuclear submarines to date and is therefore the reactor system chosen for the present analysis. We shall first give a general description of the PWR plant and review the various configurations of its major components that may be employed. We shall then consider the plant's effects upon the associated systems and the submarine's layout and structure. Finally, we make some observations on the demands of a nuclear plant in operation and maintenance.

Outline Description of the PWR

The nuclear reactor is simply a compact but powerful heat source which is normally employed to raise steam for use in conventional steam turbines. The power from the turbine may be taken via a gearbox to turn the main shaft and propeller as in FIG. 1, or may be used to generate electricity which will supply an electrical propulsion motor. In either case, turbo-generators will convert a proportion of the steam power into electricity which will supply auxiliary machines, weapons and domestic equipment. In the case of the pressurized water reactor, the temperature and pressure of the steam is severely restricted by the maximum operating temperature (240–320°C) of the water cooling the reactor—a range of temperature chosen for efficient heat transfer at operating pressures from 140 to 165 bar (2000–2500 p.s.i.). Thus the steam will be at low pressure compared with that of modern fossil-fuelled boiler practice and, depending on the type of steam generator chosen, will be either dry saturated or slightly superheated. The thermal efficiency

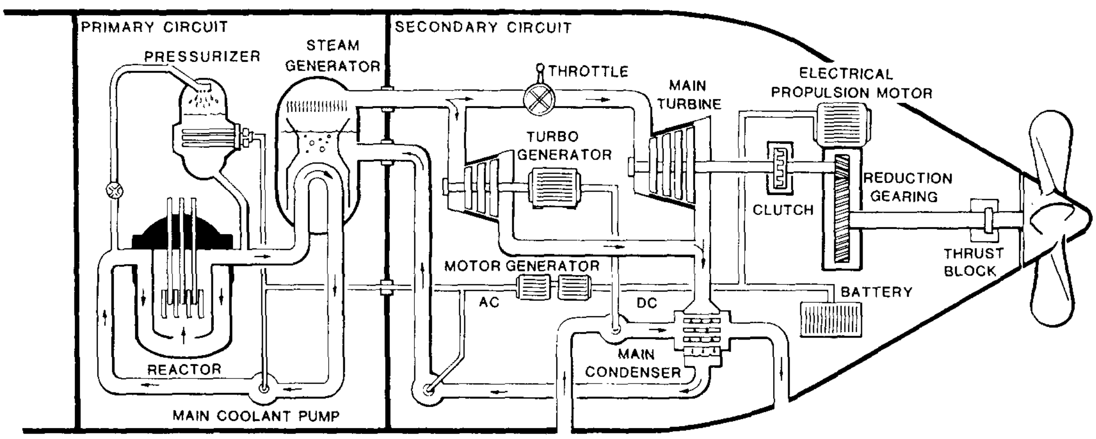

FIG. 1—TYPICAL NUCLEAR PROPULSION SYSTEM

of the plant is accordingly limited and it is found that the space requirement of the various heat transfer processes ultimately determines the maximum power output obtainable within a given size of hull.

The pressurized water is heated as it passes through the reactor core and is typically carried from the reactor pressure vessel (RPV) by large-bore pipes to one or more steam generators. This 'primary' circuit is completed by pipes incorporating the primary coolant pumps. FIG. 2 shows the principal features of a two-loop reactor plant. The numbers of steam generators and pumps are selected to give a compact plant arrangement together with sufficient redundancy to ensure high availability of propulsion power in service.

The pressurizer is a vessel connected to the primary circuit, in which the water is electrically heated above the primary circuit temperature to the saturation temperature equivalent to the desired operating pressure. The upper portion of the vessel is steam-filled, providing a cushion against volume changes of the primary coolant which occur during power transients. Automatic spray injection of cooler water from the primary circuit compensates for any excess heat input or volume surge which tends to compress the steam and produce excessive pressure. Owing to the 25% or more reduction in density of the primary circuit water between room temperature and the plant operating condition, provision is made for make-up and discharge to keep the pressurizer water level within a desired operating range.

Materials of the reactor core and primary circuit must be chosen with care for their various requirements of strength and resistance to corrosion or radiation damage. To assist in minimizing corrosion, additives are introduced to the primary coolant which is otherwise simply water of high purity. A coolant treatment system is connected to the primary circuit to maintain the desired water conditions.

Alternative Plant Arrangements

Within the general concept of the pressurized water reactor, several distinct arrangements of the plant are possible and have been employed in nuclear propulsion systems. There are two basic questions of arrangement which have to be answered. The first is whether to enclose the principal components such as the reactor and steam generator in a single pressure vessel or in separate pressure vessels connected by pipework. The second and related question is whether to adopt a recirculating steam generator with its associated

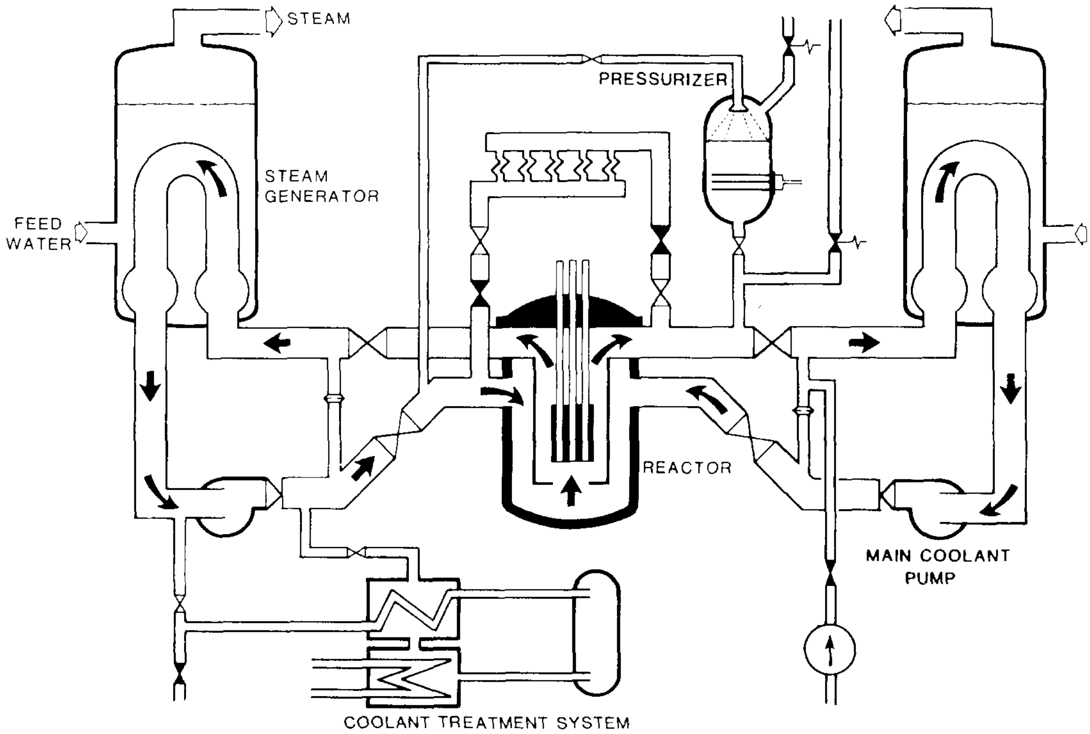

FIG. 2—SIMPLIFIED DIAGRAM OF A PRESSURIZED WATER REACTOR

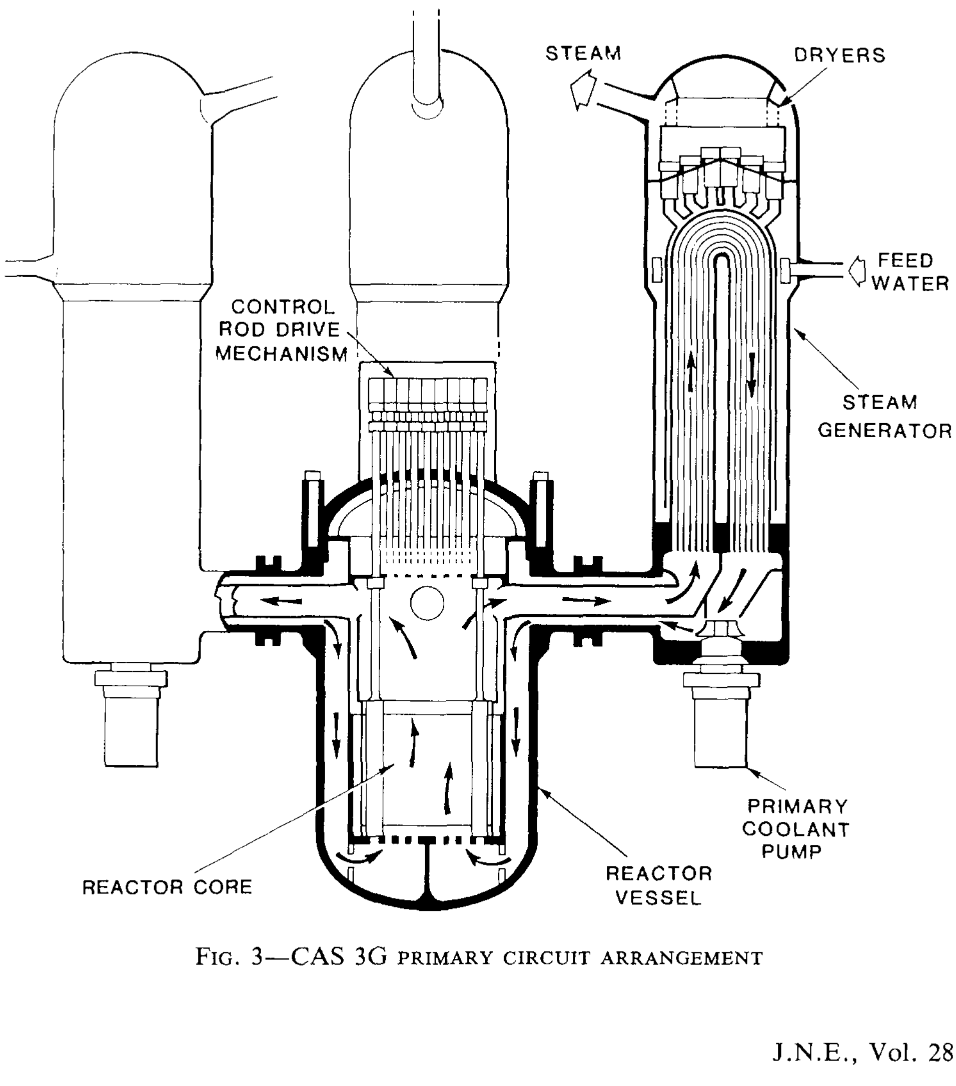

FIG. 3—CAS 3G PRIMARY CIRCUIT ARRANGEMENT

steam separation equipment or a 'once-through' steam generator. The once-through type usually, but not necessarily, carries the secondary fluid inside the heat exchanger tubes and the primary fluid outside the tubes and within the pressure shell.

The loop-type reactor, in which the principal components are separate entities connected by pipework, has been used in British nuclear submarines and in commercial ship projects such as the American N.S. *Savannah* and the more recent Japanese nuclear-powered ship *Mutsu*. Larger sizes of loop-type PWRs are now employed in many countries for land-based electricity generation.

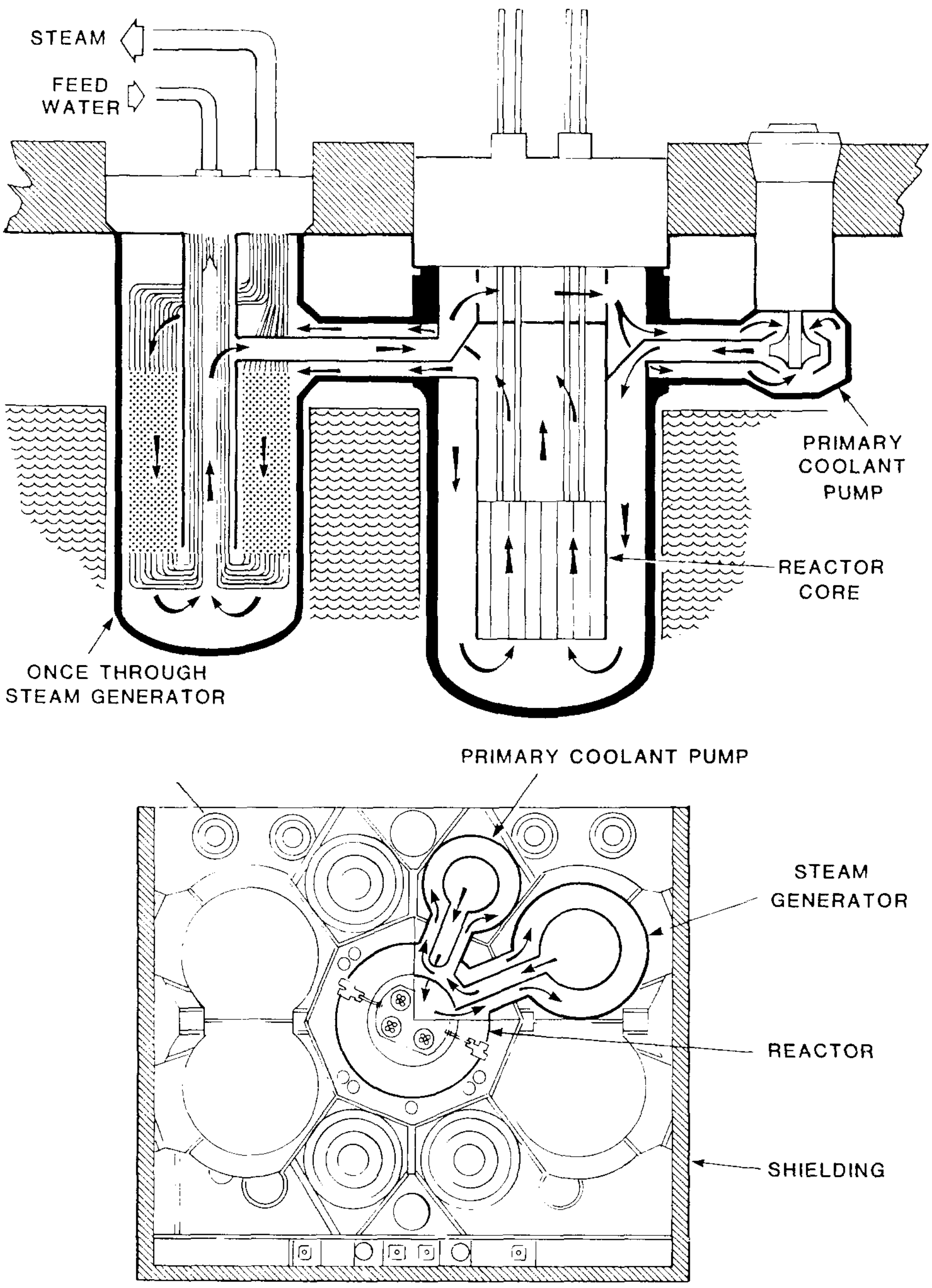

FIG. 4—'ARKTIKA' PRIMARY CIRCUIT ARRANGEMENT

A variant of the loop-type reactor is the 'close coupled' type in which the main components, while in separate pressure vessels, are brought together with only short interconnecting nozzles. Compactness and low pumping power are the incentives for such an arrangement. One version of such a plant is the proposed French Chaufferie Avancée de Série (CAS 3G) shown in FIG. 3 which has been offered for surface ship propulsion. Another is the nuclear plant of the Russian icebreaker *Arktika* which is shown in FIG. 4. In this case the four steam generators, positioned closely around the reactor pressure vessel, are of the once-through type with the primary fluid outside the boiler tubes.

An even closer coupling of the major components is achieved in the French Chaufferie Avancée Prototype (CAP) reactor (FIG. 5) in which the recirculating boiler is positioned directly above the reactor so that the tube-plate of the steam generator serves also as the reactor vessel closure head. This configuration involves some mechanical complexity but is compact and

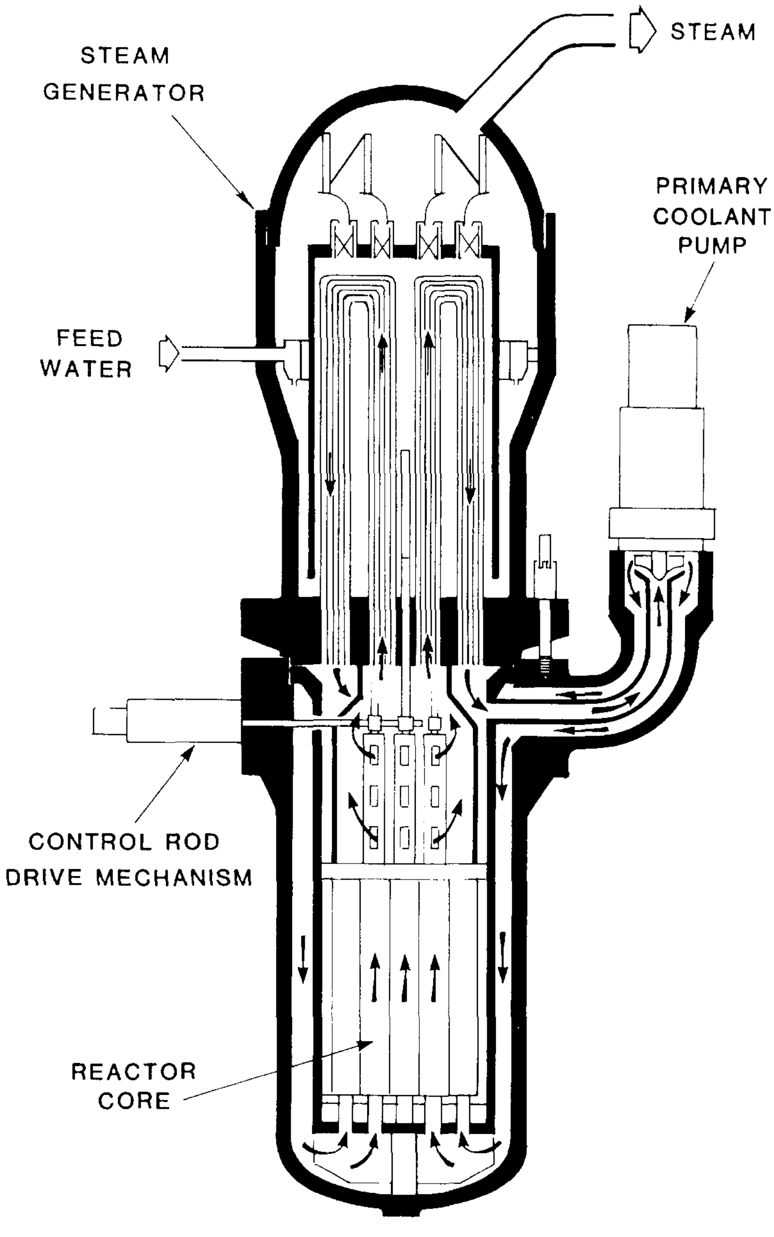

FIG. 5—CAP PRIMARY CIRCUIT ARRANGEMENT

enables the plant to be operated at lower power outputs with natural circulation of the primary coolant if the pumps are stopped. This reactor was designed and operated partly as the prototype of a new marine reactor which is reported to be employed in the RUBIS Class of nuclear submarines now being built in France.

Finally, we come to the 'integrated' PWR in which the reactor and a once-through steam generator are arranged within a single pressure vessel with primary coolant pumps mounted on short connecting nozzles. One example is the Consolidated Nuclear Steam Generator (CNSG IVA) designed by Babcock and Wilcox in the U.S.A. (FIG. 6). This developed version of an integral reactor design had once-through steam generators in a series of cylindrical pods within the main reactor vessel, with the primary coolant carried from top to bottom inside the heat exchanger tubes. Shown in FIG. 7 is the reactor which operated for many years in the nuclear ship *Otto Hahn*. It was developed in West Germany from an earlier version of the CNSG. It has the secondary fluid inside the steam generator tubes and is unusual in that the pressurizer is also incorporated within the reactor pressure vessel, there being a water surface in the upper part of the vessel with steam above.

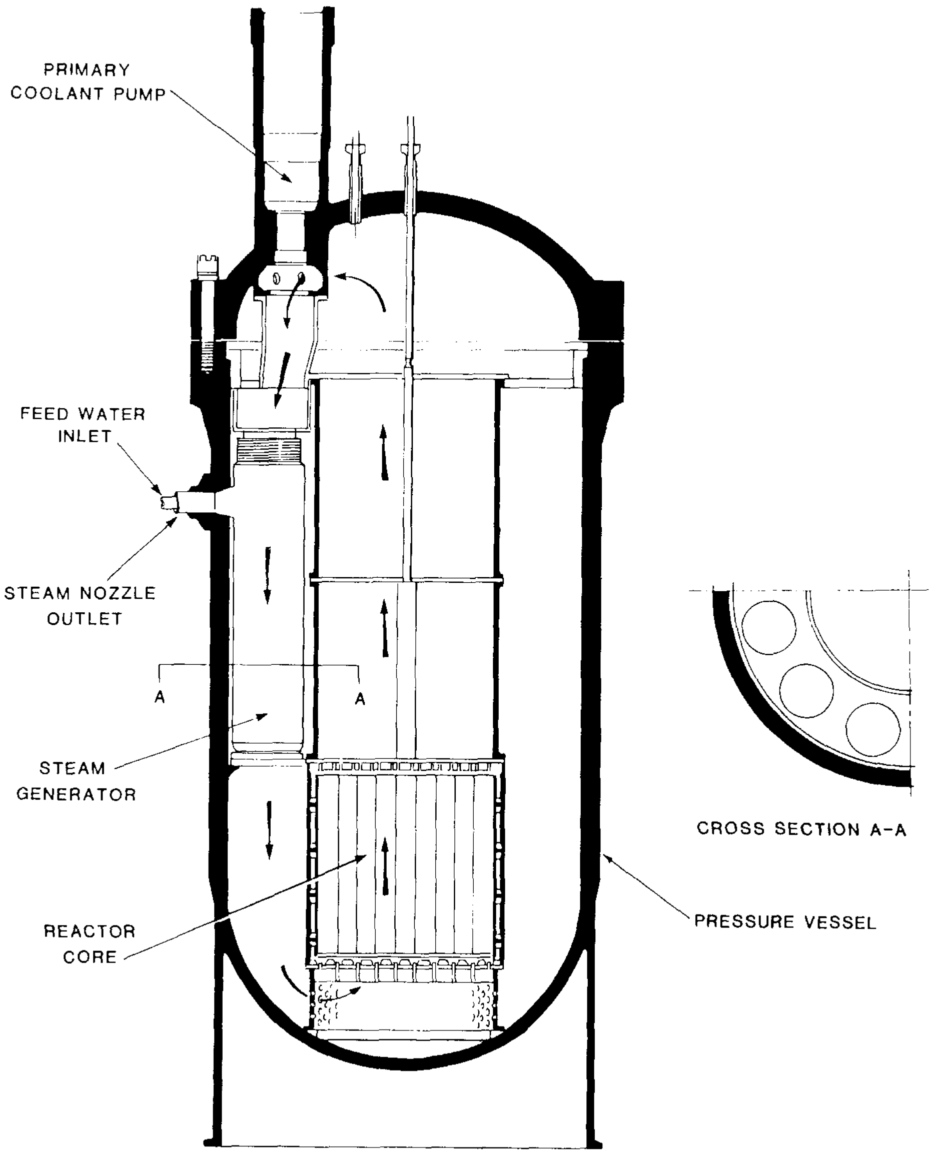

FIG. 6—CNSG IVA PRIMARY CIRCUIT ARRANGEMENT

The steam for the pressurizer arises from the hot primary coolant leaving the reactor and some boiling must therefore take place within the reactor core. All of the water in the system remains close to the saturation condition and the operating pressure is therefore well below that of the other reactor types described. Advantages of integral reactors include a weight reduction deriving from compactness and, again, the ability to produce some power with natural circulation of the primary coolant. However, the detail design must ensure that the proximity of other major components to the reactor core will not unduly complicate the maintenance and repair processes.

All of these reactor plant variants could conceivably be used, if appropriately scaled in size, for submarine propulsion. In the following sections however, the loop-type reactor is chosen to illustrate some of the principal effects of the nuclear plant on submarine design and operation.

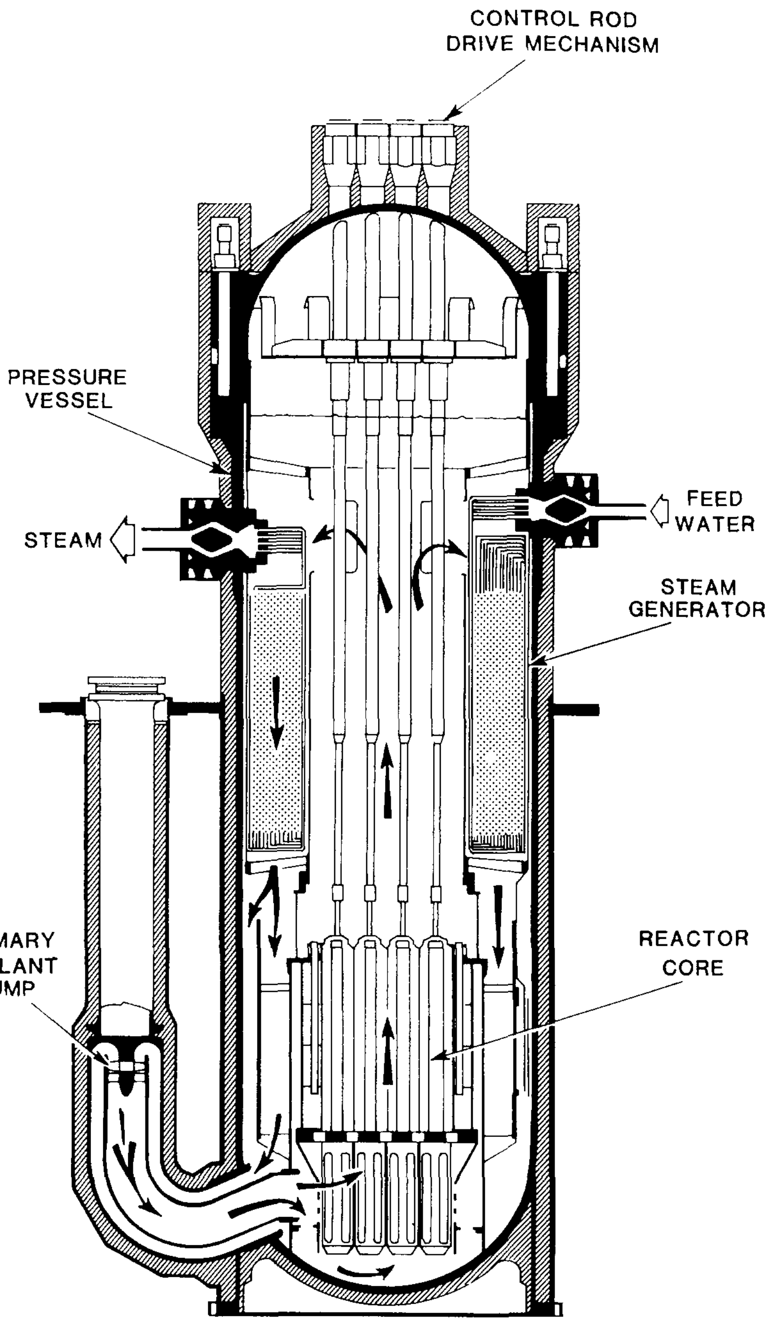

FIG. 7—'OTTO HAHN' PRIMARY CIRCUIT ARRANGEMENT

Reactor Contol

The reactor is taken critical by withdrawing control rods of neutron absorbing material, until the rate of neutron production by fission is equal to the rate at which neutrons are absorbed or lost from the system. In practice, the power level at this point may be a few kilowatts.

Heat generated by the reactor fuel may then be used to raise the temperature of the primary coolant into the normal operating band. This is easily achieved if the reactor core is designed to have a large negative density coefficient of reactivity, where reactivity is a measure of the tendency for the number of neutrons present to increase or decrease. As the neutron population of the core is increased by further control rod withdrawal, the reactor power will rise (to a megawatt or so) and raise the temperature of the coolant. The resultant reduction in coolant density causes increased leakage of neutrons from the reactor core and decreased effectiveness of the remaining neutrons in causing fission. The reactor will re-establish criticality at the original low power level, but at a higher coolant temperature. Hence, coolant temperature can be raised progressively by incremental rod withdrawal until the operating temperature band is reached.

With the reactor at normal operating temperature, full power operation becomes available. Owing to the large negative reactivity coefficient, an increase in power could be achieved without initially moving the control rods. The increase in steam flow from the steam generator causes the primary coolant temperature to drop; neutron leakage from the reactor is reduced and the core power increases to meet the demand. The average temperature of the primary coolant in the core is ultimately restored to a desired value by control rod adjustment.

The automatic response to an increased power demand permits a rapid load-following capability from the core, but may create difficulties for the designers of the propulsion machinery. Two such problems can be described. Firstly, there is a period early in the transient when steam flow and steam pressure are both high; this is a characteristic of the recirculatory steam generator with its large thermal capacity and means that the turbo-machinery must be designed to absorb the appropriate transient loads. Secondly, there is a dip in steam pressure during the transient which must be within the load-following capability of the turbo-generators.

In the case of a power-reducing transient, the reduction in steam offtake causes primary coolant temperature to rise, driving down the reactor power, which will finally match the reduced steam power. For the extreme forms of this manoeuvre it is essential to ensure that the feed and steam systems are designed to accept the consequences of a high transient steam pressure, without for example lifting the steam generator relief valves. Such problems are not intractable: they merely underline the need for matching of the reactor plant design with that of the remaining propulsion machinery. The end result is a plant which has an optimized response to load changes, being both fast and simple in operation.

Retention of Fission Products

The fission of uranium 235 in a reactor core produces a range of fission products (fragments of the fissioned atoms) and it is the absorption of the kinetic energy of these fragments within the fuel that is the major source of heat when the reactor is at power. Many of the fission products are highly radioactive, some remaining active for thousands of years. They produce radiation during their radioactive decay to more stable isotopic forms and the local absorption of that radiation generates a considerable amount of heat. For a reactor system to be safe it is imperative that the fission products are retained, since their radiation is biologically harmful.

The modern trend in nuclear engineering is to define a risk target that relates the magnitude of any fission product release to the environment to its frequency. Typically, such a target might aim to ensure that a major nuclear accident (associated with the release of most of the fission product inventory) occurs no more frequently than once in a million plant years. Such targets present a considerable challenge, not least because they require the reactor designers to contemplate sequences of events that are outside their experience in practice. No less important, as the owners of the Three Mile Island plant near Harrisburg, Pennsylvania, will testify, is the requirement to avoid core damage on economic grounds: even if an accident results in a very limited hazard to human health and the environment, the cost of clean-up and plant repair following major core damage is enormous.

The retention of fission products can be discussed in terms of the barriers that are engineered to prevent their dispersion. The three main barriers are:

(*a*) the fuel element cladding;

(*b*) the primary circuit pressure boundary;

(*c*) a 'containment' pressure boundary.

This approach is adopted below, with reference where appropriate to the effects of probabilistic targets and their interaction in the design of a submarine.

Fuel Elements

A limited leakage of fission products from the fuel is permissible in a civil reactor, where space and economic factors permit the design of complex clean-up systems and a shielding concept to match. In a submarine, because of its enclosed nature, the first objective of the core designer must be to ensure the complete retention of fission products within the fuel elements, i.e. to prevent their dispersal to the primary circuit and hence ensure that safe operational radiation levels are maintained. The attainment of such a high standard of fuel cladding integrity must obviously begin in the choice of the basic fuel element design and for this reason the design of submarine reactor fuel may well take a different path from that chosen by civil reactor designers, whose prime aim is to minimize fuel and operating costs within the appropriate safety regulations.

Once installed and operated, the integrity of the core is dependent upon maintaining effective cooling, so that the fuel element temperatures assumed in the design are not exceeded. The principal requirement is to avoid 'burn-out', a process in which rising heat fluxes lead to interruption of the normal heat transfer mechanisms and allow fuel element temperatures to increase very rapidly, potentially at hundreds of degrees centigrade per minute.

This situation is avoided by ensuring close control of reactor power and coolant conditions of pressure, temperature, and flow. It is first necessary to determine by laboratory experiments the conditions of incipient burnout and relate these to the major plant parameters by the use of computer models. Protection is then provided by designing a hierarchy of alarms, automatic control systems and automatic trips which strike a balance between the need to maintain the reactor at power whenever possible (for ship safety) and the need to prevent core damage (nuclear safety).

This process is complicated by the requirement for a rapid dynamic response in a marine reactor, which necessitates the provision of fast-acting controls (which might malfunction or be incorrectly operated) and the requirement to absorb overshoots and undershoots in major plant parameters during normal or emergency manoeuvres. It is imperative that the reactor alarm and protection systems do not function during such manoeuvring transients.

In the limit, the reactor may be shut down ('scrammed') by rapidly inserting the control rods, which quickly reduces the reactor power to a low level. A measure of the success which has been attained in the prediction of transients and the design of reliable protection systems can be drawn from the fact that the incidence of automatic shutdowns in Royal Navy plants compares well with the incidence of such events in civil nuclear plant, despite a much more severe operating environment.

The requirements for reactor protection have two major effects on the design of the submarine:

(*a*) High integrity electrical supplies are required to maintain reactor flow and to power the control and instrumentation systems. Duplication of major supplies is essential.

(*b*) Numerous sensors are required, generally in the reactor compartment, which must be processed by a complex electronic system with majority voting logic to ensure reliability and freedom from spurious trips. This system will require cables to be run from the reactor compartment, and space to be provided in an environment suitable for the electronic processing equipment.

Primary Circuit—LOCA Protection

A loss of coolant accident (LOCA) presents special problems. It will be recalled that the radioactive fission products created in the fuel have two undesirable characteristics: radiation and its associated heat generation. It is no exaggeration to say that the capacity to generate 'decay heat' presents the largest single safety problem to the reactor designer and has a significant effect upon the design of the submarine.

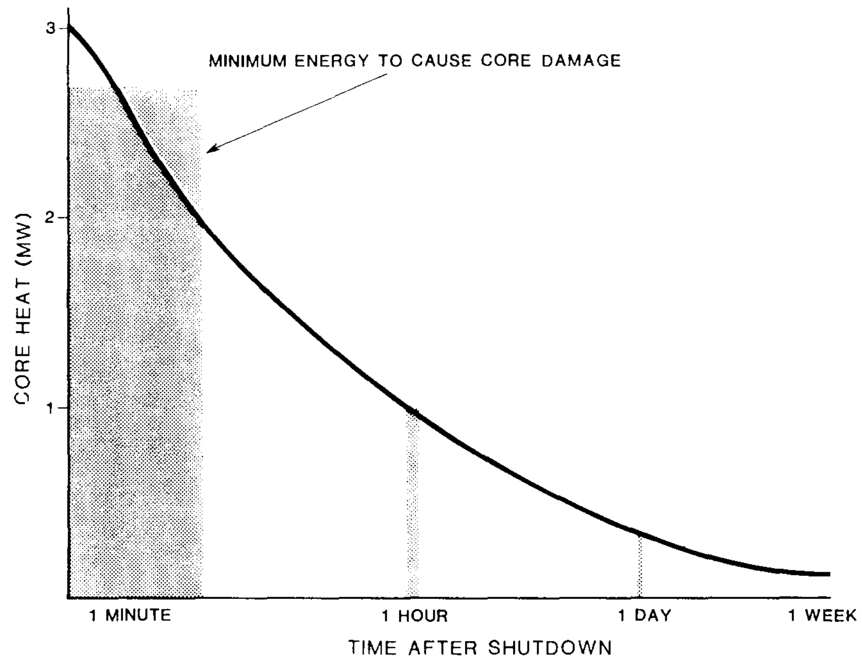

FIG. 8—TYPICAL REACTOR DECAY HEAT CURVE

FIG. 8 provides an indication of the power generated by a typical submarine reactor after shutdown following a period of high power operation. The area of the columns on the graph is intended to illustrate the heat energy required to raise the temperature of the core structure to a level at which extensive damage (and fission product release) will occur in the absence of core cooling. In energy units the area of each column is identical: they become smaller owing to the logarithmic time base. The plot is intended to show that for long periods the decay power is easily capable of causing major core damage if cooling is not provided. It is not adequate simply to shut down a nuclear reactor following an accident. Core failure can still occur. It is not even adequate to cool the core for (say) 24 hours: if cooling is then lost there may still be sufficient decay power to cause major damage. It is this phenomenon that requires the provision of protection features against the loss of coolant accident, which is the most significant single threat to a PWR.

The fundamental protection against a LOCA lies in the use of suitable materials and the development of a high integrity pressure retaining boundary. The primary circuit and major components of the plant are designed and manufactured to very high standards and are subject to detailed inspection and pressure testing.

However, loss of coolant can still occur and it is necessary to reduce the risk to a very low level. This can be achieved in two ways:

(*a*) Leak isolation. Isolating valves are fitted in each of the primary circuit loops as close to the RPV as possible, and are designed for rapid closure so that the threat to the core from a loop leak can be eliminated and power operation maintained if necessary in an unaffected loop. Isolation is also provided for all the major connecting systems. The submarine's electrical and hydraulic systems will be required for valve actuation.

(*b*) Coolant make-up. Make-up systems may be used to ensure that coolant can be added to the primary circuit following the detection of a leak. Again reliable electrical supplies will be needed.

A further requirement for decay heat removal may ultimately occur if it becomes impossible to transfer heat via the steam generators, for example following the loss of the feed system or loss of primary circulation upon closure of the main loop isolating valves. Without cooling, loss of coolant via the primary relief system would occur. In case all power supplies are lost a facility which relies on natural circulation is provided for this purpose, transferring heat ultimately to the sea. The positioning of this cooling system, which should be able to operate under angled boat conditions, may prove difficult in view of the limited height available above the level of the heat source. If such a cooling system cannot be accommodated in the reactor compartment, then its horizontal distance from the core becomes important, as well as its incorporation into the containment design.

Containment

The provision of a well-designed primary circuit boundary and of highly reliable reactor protection systems will ensure that fission product release from the plant is very rare.

However, the consequences of such a release would be correspondingly serious and it is necessary to provide a third retention boundary which is normally referred to as the 'containment'. A schematic diagram of a submarine containment is shown in FIG. 9. It is useful to describe the contribution made to its specification by the reactor plant designer and then to consider how it might be implemented in the design of the submarine. The objective of containment analysis is to define the structural and operational requirements in a manner which will satisfy a probabilistic safety target. The process has three elements:

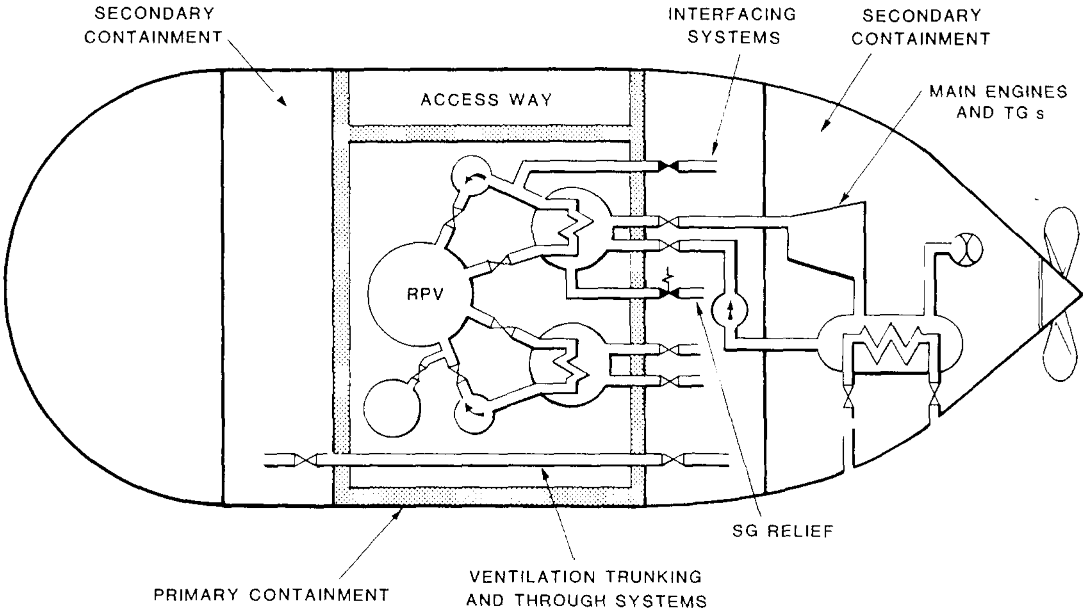

FIG. 9—SCHEMATIC DIAGRAM OF FISSION PRODUCT CONTAINMENT

(*a*) *Event Sequences* The initiating events which could lead to fission product release are defined and a prediction is made of the likely frequency and course of events following each initiator. Examples are:

(*i*) a reactor transient which causes limited fuel damage but is associated with fission product discharge through the primary relief system;

(*ii*) a LOCA which results in a major release of fission products to the reactor compartment;

(*iii*) a LOCA as in (*ii*) above, which also damages the containment structure.

The predictions for each event will concentrate on defining the pressures and temperatures within the containment and the state of the reactor fuel which, in the presence of residual water, may itself affect the physical conditions within the containment.

(*b*) *Consequence Analysis* The pressure/temperature/core condition predictions may be used as forcing functions in a mathematical model of the ship's structure, which will take into account any leakage and diffusion of fission products in the various volumes. By this means, the rate of release outside the hull can be predicted. There may be a spectrum of release possibilities for each initiator.

(*c*) *Design Policy* The analyses described above enable the reactor plant designer to develop a picture of the demands that may be made on the various structures, and of their frequency. A design and operational policy can then be evolved and tested where necessary by revised calculation. For example (referring to FIG. 9), the reactor compartment bulkheads might be designed for a particular pressure on the basis that it covers most eventualities: the event sequences leading to pressures marginally greater than this are rare and can probabilistically be expected to lie within the predicted safety margin of the structure.

Similarly, a prediction of the timescale of pressure transients and their likelihood might permit the conclusion that the access way need not be designed to the same strength as the bulkheads, provided an appropriate operational policy is defined. Further, it may be possible to make use of the potential for a 'secondary containment', i.e. a closed volume possibly weaker than the primary containment but nevertheless capable of ameliorating the effects of primary containment leakage. This might modify the leak-tightness requirements of the primary containment bulkheads.

By the process described it is possible to evolve a containment design that is acceptable to the ship designer and is demonstrably safe. It is also possible to provide guidance on other issues: for example the isolation requirements for systems penetrating the containment, the reliability requirements for various damage control systems and the post-accident operational policy. The enclosure of the reactor system ensures that the crew will be able to function in such emergencies and the strengthened bulkheads may be planned to fulfil other compartmentalization requirements.

Radiation Shielding

The final design issue for discussion is the requirement for radiation shielding. A satisfactory shield design is vital to the safety of the crew and should achieve standards of radiation attenuation which are consistent with international recommendations, despite the restricted space within the submarine. The problem is accentuated by the need for through-access at all times, the desirability of early entry to the reactor compartment after shutdown, and the continuous periods spent within the submarine by its crew.

The nature of the shield design will be strongly influenced by the type of reactor system chosen: an integrated plant may be more efficiently shielded than a loop-type PWR which requires substantial shielding dispersed on the bulkheads. FIG. 10 shows a schematic shield design for a loop-type PWR. In the optimization of this design the shield designer cannot, in general, opt for a conservative approach at each stage because of weight and space considerations. The principal design requirements of the shield are discussed below.

Of the various forms of radiation that are produced in an operating reactor system, neutrons and gamma rays of various energies are by far the most penetrating and a shield that is satisfactorily designed against them will achieve the required attenuation for other forms of radiation. Their absorption and interactions within the shield materials are complex and the following paragraphs define the major requirements during normal reactor operation.

Primary radiation (emitted directly from the core) consists predominantly of fast (high-energy) neutrons and gamma rays from fission and from the decay of fission products. Some attenuation of this radiation may be achieved within the reactor pressure vessel by the use of a steel thermal shield whose prime purpose is to help control the radiation exposure of the RPV. Neutrons leaving the RPV are most efficiently slowed down and captured by the process of elastic scattering in a material of low mass number, a particularly good example being 'bound' hydrogen. Water is excellent for this purpose and a shield can be arranged by surrounding the reactor vessel with water in a primary shield tank, as shown in FIG. 10. The upper surfaces of the RPV, which might require periodic access, may be shielded with polyethylene (polythene) blocks which have similar shielding characteristics. As the design of the shield is developed, careful attention must be paid to the temperatures generated within its structure and to the pipework which might permit radiation streaming.

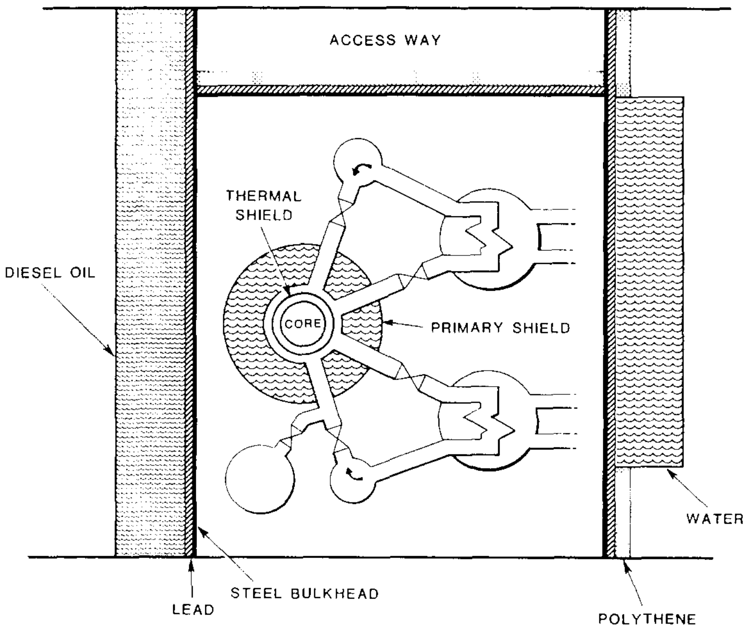

FIG. 10—SCHEMATIC DIAGRAM OF RADIATION SHIELDING

A major objective of the primary shield design is to ensure that radiation levels are reduced sufficiently to permit entry to the reactor compartment very soon after shutdown: in common with civil PWRs, the neutron and gamma radiation levels within the compartment would prevent access during power operation. Acceptable levels outside the reactor compartment are achieved by the combined design of the primary and secondary shield.

In the design of the secondary shield maximum advantage is taken of reactor plant components and structures which must exist for other purposes in the boat, such as water and diesel oil tanks. Where necessary, polythene may be incorporated on the bulkheads and access way to attenuate further the neutron radiation.

In addition to the gamma radiation emanating from the core, secondary gamma radiation will be produced from at least three sources:

(*a*) The primary shield array as a result of neutron capture in all materials, which become either instantaneous radiation sources or activation products that decay subsequently with a characteristic half-life.

(*b*) The primary coolant, as a result of the reaction of fast neutrons with oxygen atoms passing through the core. This reaction produces nitrogen-16, which has a half-life of 7 seconds and which emits a high energy gamma ray during decay. This gamma source will extend throughout the primary circuit.

(*c*) Irradiated corrosion products, which may be locally distributed around the primary circuit and concentrated within certain systems, for example in ion-exchange columns which may act as filters.

Of these sources, (*a*) and (*b*) are of prime importance for shielding during operation while (*c*) can dictate shutdown radiation levels within the reactor compartment.

The attenuation of gamma radiation requires a substantial mass per unit area of shield. To minimize the space occupied, the material should be of high density and lead may be selected for this reason and for its ease of fabrication to provide shielding of both the reactor compartment bulkheads and the access way.

The shielding forms a significant fraction of the total weight of the nuclear equipment. As in the case of the primary system itself, such heavy weights within the submarine must be distributed with boat stability in mind. We may thus expect the reactor compartment to be sited centrally in the boat's length and heavy components to be kept as low as possible. Despite these features, the nuclear propulsion plant does not compare too unfavourably in total weight with conventional plant, especially when the latter's fuel inventory is taken into account. Of course, when making such comparisons it must be remembered that the nuclear plant offers a high speed performance and sustained submergence capability with which the conventional plant cannot compete.

Operating factors

As we have seen, the levels of neutron and gamma radiation require that personnel be excluded from the reactor compartment during operation. A significant consequence of this is that the crew no longer enjoy that direct access for local manual control of valves and other equipment which typifies the method of working in conventional submarines. The inaccessible machinery must function with very high reliability, and monitoring instrumentation and equipment for remote control must be provided.

From his remote position the operating engineer needs to establish and maintain a clear understanding of the operating state of the nuclear steam-raising plant. It is not to be expected that an operator will respond to unexpected changes of plant condition only by peforming one or two of a small number of pre-established drills. Even if such elementary responses were desirable, human nature would ensure that a mental assessment of the plant's demands would be made and action taken accordingly. Thus the presentation of information to the operator must be sufficiently complete but not confusing. The designer must give the same kind of attention to the plant's interaction with the operator as he naturally gives to the physical interaction between the various pieces of machinery.

The automatic reactor protection systems previously described should respond to those events in which plant conditions deteriorate rapidly, thereby removing from the operator the burden of fast and accurate interpretation and response. The operator should be called upon for only limited action within the first few minutes of any failure, allowing time for a proper appreciation of the situation. As in the case of the aircraft pilot the operator must ensure that he does not concentrate on protecting the propulsion plant components from possibly minor damage when continuity of propulsion power is essential for the safety of the vehicle. The optimum response to a plant failure may therefore depend upon the submarine's operational state and fully automatic protection for all conceivable circumstances is both undesirable and impracticable.

Potential benefits from the need for central control are twofold. First, the operator is permitted a better understanding of the condition of all the machinery. He therefore develops a better mental model of its responses to his actions—a model which should be implanted and reinforced by effective training on a plant simulator. Second, central control creates the potential for reduced manning if the required operator functions are carefully analysed and accommodated. However, manning levels must always allow for the needs of on-board maintenance of equipment outside the reactor compartment.

Maintenance

Having constructed a product of initially high quality, it is necessary to ensure its reliability through long periods of service. Regular on-board testing of equipment is carried out to an extent which is compatible with the demand for continued power output from the plant. Thus, by virtue of its built-in redundancy the reactor protection system can be tested, section by section, to demonstrate that it remains functional.

Other components can only be effectively checked when the plant is shut down in harbour. Taking again the example of the primary circuit pressure boundary, a need for inspection arises from the potential growth of fatigue cracks as a result of stresses induced principally by temperature transients. Rigorous pre-service inspection is therefore supplemented by 'in-service inspection' of selected regions using similar techniques. Such examinations are often carried out during major refits when access to the plant is easier and residual radiation levels in the reactor compartment are lower. Good design and careful operation combine to minimize radiation levels during maintenance periods and, to make radiation doses to maintenance personnel as low as reasonably achievable, remotely operated inspection equipment has been developed. Automatic equipment for maintenance and repair, involving for example metal cutting and welding, also helps to keep radiation doses down. Access of personnel to radiation areas must be controlled and the adoption of nuclear propulsion brings with it the need for health physics support in fleet bases and dockyards. The refuelling of the reactor by replacement of the reactor core is required at intervals of several years. This is a specialist operation involving thorough preparatory training of personnel. In common with all other maintenance operations on the reactor plant, it must be fully controlled and documented. A significant factor in the successful management of nuclear plant repairs has been the joint responsibility for it of the Procedure Authorization Group (PAG). As illustrated by the 'safety triangle' of FIG. 11, the PAG consists of three members—one each from the repair authority, the operating authority, and the design authority. All procedures for inspection, repair, and subsequent testing must be agreed by all three parties before they are implemented. The signed approvals emphasize the responsibility of the individual members and of the organizations they represent. The PAG must also signify when an operation has been completed to its satisfaction. Independent nuclear safety assessors will normally witness the group's acitivites. A form of control such as this is essential if the condition of the plant is to be maintained to the design intent throughout its life. Recognizing that the overseeing of repairs is as important as that of the

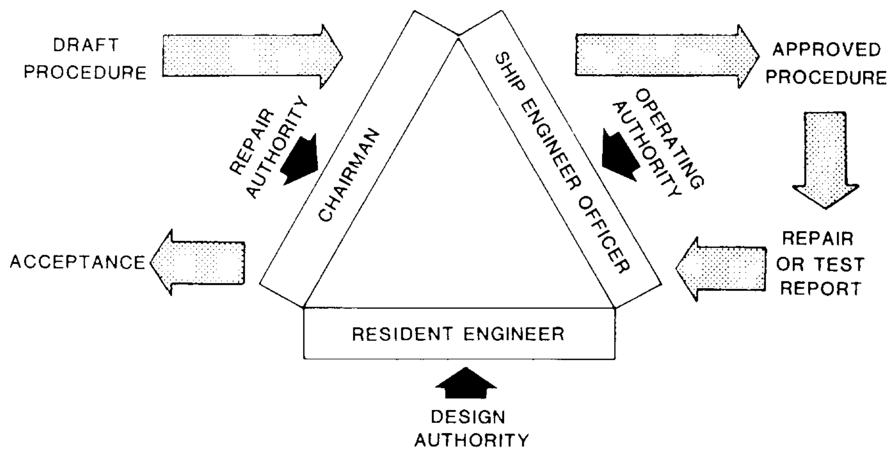

FIG. 11—MAINTENANCE AND TEST PROCEDURE AUTHORIZATION

original build (which has unquestionably made special demands upon the shipbuilder), dockyards and fleet bases must adapt their methods of working to embrace such management procedures.

Conclusion

We have described some of the essential features of the pressurized water reactor which has been commonly employed in nuclear submarine propulsion. The PWR may exist in a number of different configurations with various advantages and disadvantages, but fundamental features of nuclear power such as radiation, radioactive fission products and their decay heat all affect in some way the submarine structural design. Design optimization of both submarine and reactor plant must allow for this interaction.

Operation is affected by the inaccessibility of the loop-type reactor system for manual actuation or repair when at power. The need for high reliability and nuclear safety influences all stages of the submarine's life and makes similar demands during maintenance and repair to those in the initial design and construction stages. The extra attention to detail is well rewarded by a propulsion system of high performance and endurance.

References

1. Horlick, E. J.: Submarine propulsion in the Royal Navy; *Journal of Naval Engineering,* vol. 27, no. 1, June 1982, pp. 1-22.
2. Le Heiget, Y. and Raoul, J. P.: The Operational and experimental background of CAS reactors; *Nuclex* 1978, Paper C1/5.
3. Mel'nikov, E. M. et al.: Experience acquired from the development of the nuclear propulsion unit of Ice-breaker 'Arktika'; *Sudostroenie,* no. 1, 1977.
4. Anon.: Two designs for small and medium capacity LWR power generators; *Nuclear Engineering International,* vol. 20, Jan. 1975, pp. 51-52.
5. Anon.: A Small pressurized water reactor for process energy—plant costs and design studies; *Babcock and Wilcox Company Report* BAW 1428, ORNL-Sub-4390-2, June 1976.
6. Anon.: N.S. 'Otto Hahn'; *Directory of Nuclear Reactors,* International Atomic Energy Agency, vol. 9, 1971.