Zeitschrift: IABSE congress report = Rapport du congrès AIPC = IVBH Kongressbericht

Band: 6 (1960)

Artikel: Various problems in the design and construction of nuclear reactor structures

Autor: Billig, Kurt

DOI: https://doi.org/10.5169/seals-7006

VI 2

Various Problems in the Design and Construction of Nuclear Reactor Structures

Problèmes divers que posent l'étude et la construction des centrales nucléaires

Verschiedene Probleme bei der Projektierung und beim Bau von Atomkraftwerken

KURT BILLIG
Prof. Dr.-Ing., M.I.C.E., M.I. Struct. E., M. Am. Soc. C.E., M.I.C.E.I.
Great Britain

It is not intended to discuss here such problems of power reactor design as achievement of chain reaction, stability and control, neutron economy, fuel technology, coolants, or radio-activity. These and similar subjects concern mainly the physicist, the chemist and the metallurgist. Neither is it intended to touch on problems connected with heat removal and heat transfer, power production, reactor control, or instrumentation. These are questions dealt mainly by the mechanical and electrical engineer. The present Paper is intended to give a summary survey on the various problems in the design and construction of nuclear reactor structures, that is, it will deal only with subjects which are the concern of the civil and structural engineer. Some of the principal problems discussed here deal with siting, containment, location of reactors underground, foundations, pressure vessel, reactor shielding, engineering considerations, research and development.

Siting

Some of the main factors in the siting of nuclear power stations are: the distance from the place of power requirements, a good supply of cooling water, good load bearing soil to carry the heavy loads of reactor structures, considerations of safety and public opinion on the hazards of nuclear radiation.

The three basic principles of protection from nuclear radiation are: distance, time and shielding. The influence of distance is given by the inverse square law caused by radiation in all directions. Considerations of time may be due to the decay of the active material, or by restricting the time of exposure. Shielding as a means of protection must not be restricted to direct beams of radiation, but also take account of scattering. The degree of success of the protective measures taken should always be assessed by a stringent control system.

Containment

If a failure of the emergency shut-down system be postulated and a considerable excess of heat release over heat removal (caused by sudden rise of power or failure of cooling), the reactor fuel elements may melt and possibly release a radioactive cloud of fission products into the air. This cloud may drift downwind and contaminate a considerable area.

Although a double fault of the kind postulated is highly unlikely it cannot be entirely rejected as a possibility. Since the consequences of a serious accident in a populous area would be severe, reactors have either to be sited in nonpopulous areas or the less safe reactors provided with an external container which will prevent any significant escape of activity in the event of the most serious accident which could occur. The design of such reactor containers presents many interesting problems.

The magnitude of credible potential nuclear excursions is difficult to estimate, because any credible combinations of circumstances that might lead to a nuclear excursion will be specifically designed against.

However, disregarding consideration of any specific mechanisms that could lead to such an excursion, it appears reasonable to suppose that the excursion energy would not exceed that which would be just enough to melt all of the uranium fuel in the core. This energy may be considered as the maximum reasonable nuclear contribution to the post-accident internal pressure.

It is not suggested that melting of the entire core would be the actual mechanism whereby the core is disrupted, thereby terminating the nuclear reaction. Vaporisation of the highest-neutron-flux (central) portions of the core while the lower-flux regions are still in a solid state appears to be a more likely mechanism of disruption. The full-core melting model merely provides a convenient way of expressing a reasonable upper limit to the magnitude of a potential nuclear-excursion energy contribution.

Setting the design basis for a reactor enclosure represents a task quite unique in engineering experience. For here we are dealing with a structure designed for a function that it will almost certainly never be called upon to fulfill. This surrounds the specification of that function with quite unique uncertainties.

The containment vessel will provide highly reliable assurance that the public will not be afflicted with a disaster even in the unfortunate event that the extremely improbable should happen.

A concern of the atomic industry in designing such enclosures is that, in view of the uncertainties surrounding the functional specifications, these should not be set with unrealistic pessimism. The direct and indirect costs of an enclosure represent a substantial contribution to the total cost of nuclear power plants, in addition to imposing inconvenient restrictions on plant layout.

In an attempt at finding a compromise solution between the costly provision of a separate containment vessel and the alternative extreme of making no provision for containment at all, the practicability of semi-containment is now being considered and discussed. According to this proposal, the biological shield shall be capable of being effectively sealed against a nominal pressure within a period of a few minutes, with a view to containing fission products subsequent to a pressure vessel failure.

The Location of Reactors Underground

At various occasions the positioning of reactors underground has been discussed and there is at least one project where an experimental reactor has actually been placed in solid rock, namely, at Halden, Norway. There is also a Swiss project, still in the design stage, in which the reactor, the primary heat exchangers and auxiliary reactor equipment are placed in one cavern while all secondary equipment, turbines, heat plant, electric supply and distribution plant, and control room are placed in a second cavern.

The main reasons for giving serious consideration to the underground location of reactors are probably three: first, the matter of containment; secondly, safety from the point of view of military operations; and thirdly, the appearance of the landscape after the power station has completed its service.

With regard to the problem of containment, it is believed that in solid rock satisfactory explosion resistance can be obtained. Solid rock is reasonably airtight and the inside rock faces of the hall can be lined with concrete and cracks in the rock firmly sealed. Any minor leakage which may occur will be considerably delayed if the rock cover above the reactor hall is of the order of 100 ft. or more. A thin gas-tight seal, in the form of a steel plate, can be added inside the hall if found to be necessary at a later date.

To satisfy the military safety of a nuclear power station, it would be necessary to base the whole station underground, including the control room — a layout which has been followed in the Swiss project.

From the point of view of the appearance of the site, it would be only necessary to place underground the reactor itself, while the steam-raising plant, turbines and ancillaries may be positioned above ground in the usual way. This layout is likely to result in the minimum additional cost.

In Norway, where excavation techniques in rock are highly developed, it has been found that the underground containment of a reactor differs very little in cost from that of a conventional building, and is considerably cheaper than the provision of a separate steel containment such as a spherical shell.

Foundations

A major factor that governs the design of reactor buildings is differential settlement, in particular any movement between the reactor itself and the gas blowers and between the gas blowers and the heat exchangers. Whereas the gas ducts are fitted with bellows to accomodate movement, the amount of movement that bellows, designed for these temperatures and pressures, can in fact accomodate is not large and is mostly absorbed by the thermal movements of the ducts themselves.

In the case of Berkeley the balance of movement available to the structural engineer for differential settlement is a total of $1\,^1/_2$ inches with a maximum settlement of the reactors relative to the blowers of $^3/_4$ inch. From soil mechanics reports and other information the engineers were satisfied that the differential settlement would lie within the limits prescribed by the bellows. Actual readings of settlement during construction show that the average settlement under the main reactor raft is 0.48 inch, and that the differential settlement does not exceed 0.41 inch up to November 1958. Nevertheless, it was decided to include in the design of the foundation for a settlement joint between the main raft under the reactor and the foundation supporting the blowers and the boilers.

Irradiation Effects on Metals

Broadly speaking, the abnormal conditions imposed on metals and alloys under nuclear reactor conditions are the effects of irradiation and, in some parts, unusual corrosion effects.

Most determinations of *creep* strength characteristics have shown little effect of irradiation. Experiments over a limited range have shown either no change or a slight decrease in creep strength after irradiation. However, the long-term creep resistance of high temperature alloys working near their upper limits of creep conditions have yet to be fully determined under strong irradiation.

Most metals become *hardened* by irradiation and the hardening may persist after the metal has been heated to temperatures at which most of the irradiation damage, as shown by other properties, has been annealed out. Many of the properties of metals which are changed by irradiation can be restored by annealing. However, irreversible damage such as accumulation of fission products in fuel metals, creation of new elements, physical cracking, or dimensional changes are not affected by annealing.

The general effect in soft metals is that irradiation *raises the elastic limit* considerably and reduces the rate of strain hardening. The yield stress in copper at room temperature is raised from almost nil to an appreciable figure by irradiation and the elongation to fracture is reduced.

Neutron irradiation may increase the tendency of a metal to fail with a *brittle fracture*. Such loss of ductility can be serious, particularly for instance, in the design of reactor pressure vessels. The embrittlement effect is most easily studied by measuring the effect of irradiation on the transition temperature of the metal, above which it will be ductile and below, brittle. In general, the effect of irradiation is to raise the transition temperature, thus giving a specimen a greater chance to fail by brittle fracture at room temperature, say. The influence of irradiation is markedly affected by temperature of irradiation and embrittlement effects are less at high temperature. This is again supported by the fact that it is possible to reduce the tendency to brittle fracture, after irradiation, by annealing.

Mild steel at normal temperatures is ductile, but at temperatures below the range $-40°$ C to $0°$ C behaves as a brittle material. Research on certain alloy steels has shown that after a prolonged period of irradiation embrittlement may occur at temperatures perhaps $60°$ C higher than this, i.e., within the normal working range. A considerable amount of research is now being carried out to investigate the risk of embrittlement of mild steel under prolonged and strong irradiation.

Resistance of Concrete to Radiation Damage

An early indication that radiation damage in concrete is not very serious was given by the behaviour of the Oak Ridge reactor concrete shield, which, though unprotected by a thermal shield, suffered no apparent deterioration during the course of $5^1/_2$ years' operation. During this time the integrated dose of radiation at the shield surface amounted to 3×10^8 r and 10^{18} neutrons/cm^2. This behaviour was confirmed by a series of careful measurements carried out during 1953 to 1956 at Harwell. Forty-eight concrete specimens 2 in. $\times$ 2 in. $\times$ 8 in., suitable for strength determination by transverse rupture, were prepared in a 1:3 Portland cement and $^3/_8$ in. aggregate mix. Some of these were broken as received, some were irradiated in the BEPO reactor, and some were kept as age controls. The blocks were irradiated in a thermal flux of about 10^{12} neutrons/cm^2/sec, the fast flux being equal in magnitude; the gamma dose rate was about 10^6 r/hr. The total energy deposition was approximately 0.01 watt/cm^3 of concrete (density 2.2 g/cm^3). If it is assumed that the total energy deposition is correlated with the radiation damage suffered by the materials, then by knowing the energy deposition per cm^3 in any given concrete shield, the probable radiation life may be deduced from the data given in the table which follows.

Table 1. Effects of Irradiation on the Rupture Strength of Concrete

a) Blocks as received: Transverse rupture: 790, 920, 930, 940 and 1000 lb./in².

b) Non-irradiated blocks, oven-tested for five months and subsequently checked for weight-loss and broken:
2 blocks at 50° C, 1030 and 1220 lb./in², 2.2% weight loss.
2 blocks at 100° C, 1030 and 1180 lb./in², 3% weight loss.

c) Irradiated blocks, irradiation temperature approximately 50° C.

Block	Time of irradiation months	Thermal flux $10^{12} \times n/$ cm^2 sec	Integrated thermal flux $10^{19} \times n/cm^2$	Total rate of energy deposit. watts/cm^3	Weight loss per cent	Rupture stress lb./in²
Q	2	1.1	0.5	0.011	2	1073
R	2	1.1	0.5	0.011	2.1	1076
S	6	1.2	1.6	0.012	2.4	918
T	6	1.2	1.6	0.012	2.6	810
U	12	1.3	3	0.013	2.2	810
V	12	1.3	3	0.013	2.6	940
W	24	1.4	7	0.014	—	734
X	24	1.4	7	0.014	—	627

(Observations by Chisholm-Batten)

These results suggest that there was no significant change in strength during the first year of irradiation (integrated thermal neutron flux approximately $3 \times 10^{19} n/cm^2$), though the two-year blocks show what might be a significant decrease. There is no evidence of the loss by irradiation of significant quantities of water of hydration. It may be inferred from these figures that, for the majority of the reactors so far built, it is extremely unlikely that there would be any appreciable change in the properties of the shield due to radiation damage during the operational lifetime of the reactor. Radiation damage to concrete is, in fact, a less serious problem in practice than over-stressing due to nuclear heating. In this respect concrete is markedly superior to organic materials.

Pressure Vessel

In determining the thickness of the vessel allowances are made for nuclear heating, for corrosion, for the temperature rise and temperature gradients resulting from heat generation in the vessel, due to irradiation effects and heat transmitted to the shell by radiation, convection or conduction and, finally, for transient temperature effects and conditions during graphite annealing. The efficiency of the joints is assumed not to exceed 90 percent.

Particular attention is given to the influence of the temperature of the vessel and pressure within the vessel on the support system and on the movement of the branches within the shielding system. Equally, the effect of the support system or other attachments on the pressure vessel is considered, and the safety of the whole system proved.

After erection the main vessel is stress relieved and this operation also covers all attachments and fittings welded to the vessel. Finally, the vessel is subjected to a pressure test in accordance with the requirements of Section 5 of B.S. 1500: 1949. In the instrumentation of this test particular attention is given to the region of the support brackets and openings in the vessel.

A pressure vessel made of prestressed concrete was used for the French reactor at Marcoule. The vessel is a horizontal 10 ft. thick concrete cylinder closed at each end by a spherical cupola with its concave face turned towards the outside. The service pressure of the vessel is 15 ats. and the proof load for the acceptance test was specified as 30 ats. The prestressing units consist of cables each designed for a working load of 1200 tons, such as have frequently been used in dam construction.

The basic point of this proposal appears to be the use of the concrete thickness which is available for shielding purposes as a storage of strength by means of prestressing. Three elements of the reactor structure are mainly involved: the Pressure Vessel, the Thermal Shield and the Biological Shield, and these three parts are interdependent from the points of view of radiation, temperature and strength. In any new structural arrangement of these parts, the distribution and relation of all the three items are bound to change. Thus, even if a new arrangement may look extremely promising as from the point of view of strength, it may not be so from the point of view of temperature and radiation; that means that every new arrangement must be checked for these three factors: strength, temperature and radiation.

The prestressed concrete vessel is quite different from the conventional steel pressure vessel in many respects: in the type of steel used, in the method of its application, in the method of construction, and in the degree of safety provided. The cables consist of the usual 0.2 in. dia. high tensile wire of 90 ton ultimate strength and minimum ductility of 3 percent. To assure gas tightness of the vessel the concrete cylinder is lined with a welded steel sheet which forms a permanent formwork and is tied back into the concrete by numerous anchorages welded to the sheet.

The structure is designed for a proof pressure of 30 ats. internally. It has also been checked for the various stages of overloading. When the excess load has increased substantially beyond the proof load, the tensile resistance of the concrete is brought into play. Although the unit strength is relatively small, the total tensile force resisted by the concrete is quite considerable in view of the great thickness of the vessel. Taking into account a tensile resistance of, say, 600 lb. per sq. in., the concrete vessel will not start cracking before the internal pressure exceeds the proof pressure by 50%, i.e., at 300% of the design pressure.

Reactor Shielding

The unique factors in a nuclear plant, which have the strongest influence on design and arrangement, stem directly from the fact that the nuclear process produces ionising particles and rays. This radiation can be biologically harmful to man, and shielding protection is therefore required.

The outward radiation from the core of a thermal reactor comprises a current of fast neutrons, a current of thermal neutrons, and a current of gamma photons. The duty of the reactor shield is to slow down the fast neutrons to thermal energy, by collision with atoms of light elements; to absorb thermal neutrons; and to absorb gamma radiation, for which purpose heavy elements are necessary.

Shielding around the reactor, gas circuits and other active plant must be designed to attenuate the radiation sufficiently to meet the specified requirements under all conditions. Particular attention must be paid to control radiation leakage at openings in the shield provided for cooling ducts and services.

For health reasons, the amount of radiation or "dose" that the human body is permitted to receive is limited. The maximum rate of dose permitted is known as the "maximum permissible level" or m. p. l. In the case of gamma radiation, one m. p. l. is 7.5 mrem per hour. In the case of neutrons, the flux corresponding to a dose rate of one m. p. l. is taken as 1200 thermal neutrons per cm^2/sec, or 30 fast neutrons per cm^2/sec, respectively.

A considerable amount of heat is generated within the shielding material by radiation in various ways: When a fast neutron is slowed down by scattering, part of its kinetic energy is transformed into heat and imparted to the nuclei of the shield. When a thermal neutron is captured in the shield, a gamma ray is liberated, and the absorption of this gamma ray releases energy, again in the form of heat.

Concrete of some form is widely used as the main biological shield in civil reactors. The temperature rise which can be permitted is thought to be about 30° C, which would correspond to an energy current entering the main shield of about 20 milliwatt per cm^2. For typical power reactors, this figure would be exceeded by something like a factor of 5, and it is therefore necessary to introduce in front of the main shield a "thermal" shield. The thermal shield reduces the energy current into the biological shield to an acceptable level by suppressing thermal neutrons and gamma currents; however, it will have little effect on the fast neutrons.

In addition to possessing high density, a high melting point, high atomic number, and a large neutron capture cross-section, a thermal shield should also be stable under irradiation, of good thermal conductivity, and cheap and easy to fabricate. Iron (in the form of steel or cast iron) was for long regarded as the obvious choice, since it possesses the required nuclear properties, and is easily incorporated into the mechanical structure of the reactor. A large number of substances could be used as neutron absorbers, but in practice the choice rests between boron- and cadmium-bearing materials.

Concrete is by far the most widely used material for the main biological shield. Its popularity is due to its cheapness, to its satisfactory mechanical properties, and most of all to the fact that it possesses many of the physical attributes of an ideal reactor shielding material. It is a mixture of hydrogen and other light nuclei, and nuclei of fairly high atomic number. It is therefore efficient both in absorbing gamma rays and in slowing down fast neutrons by elastic and inelastic scattering; and the hydrogen contained in the water of hydration of the set cement is sufficient for the rapid thermalisation of the intermediate energy neutrons. Its density can be controlled within wide limits, and it lends itself easily to monolithic construction, without the presence of heterogeneities and voids. Its principal disadvantage is the low value of the thermal conductivity which makes it difficult to extract the heat evolved in the shield as a result of the attenuation of the radiation; for a high-flux reactor the resulting temperature gradients can be large enough to constitute an important design problem.

A great deal of work has been done on conventional and special shielding concretes, and a wide variety of compositions are described in the literature. From a practical standpoint the following facts are important: Gammas, rather than neutrons, generally determine the shield thickness, even in high-density concretes. Shield thickness can therefore generally be reduced proportionately to the increase in concrete density; composition is usually unimportant. As far as the costs of materials for concrete are concerned, local variations of price are liable to be important; but superimposed on these variations is a general tendency for shields to become progressively more expensive as their density increases.

Structural concrete should not be exposed to radiation exceeding a flux of about 2×10^{11} MeV/cm^2/sec. in order to ensure acceptable thermal stresses. This flux is about 100 BTU/hr/sq. ft. and would cause about a 30° C rise in the shield. Shielding should then reduce this radiation 10^8-fold to biological tolerance level which has been defined above.

For preliminary design, heat generation in a concrete shield may be assumed to be caused entirely by gamma rays that are absorbed. H. S. DAVIES has shown that the differential equation describing the temperature distribution caused by this heat generation is similar in form to the differential equation for determining loads, shears and moments in a beam having a length equal to the wall thickness. In this analogy, loads, shears and moments along the ficticious beam correspond to the heat generation rate, heat flux, and temperature distribution through the wall, respectively. The method can also be used for determining the most effective way for cooling a shield, since the heat flux removed at a sump is treated as a concentrated load acting upward on the ficticious beam.

The determination of the thickness of shield required is a laborious calculation. Removal and diffusion theory are used to determine the distribution of fast and thermal neutrons in the shield, and hence the currents leaking from the shield. The distribution of gamma radiation from the core and from (n, γ) capture in the shield is then determined, and hence the leakage gamma current. This analysis has to include the thermal shield, and is repeated until the thickness required for the desired radiation level of fast neutrons, thermal neutrons and gamma radiation at the shield surface is obtained.

Engineering Considerations

Shrinkage cracks that can be tolerated in high-quality concrete work — those which do not affect the structural strength or the æsthetic appearance of the finished work — are not such as will cause any hazard from the point of view of radiation leakage, especially as such cracks are hardly ever along a straight line.

It seems as though specifications have in the past been too tightly drawn, and that some relaxation could be made in the requirement for uniformity of density. This conclusion is confirmed by calculations, based on the assumption that an infinite plane isotropic emitter was shielded by an infinite plane slab shield, having an attenuation factor about equal to that of a reactor shield, i. e., approximately 10^8. The calculations assumed that the radiation travelled in a straight line, and that it was exponentially attenuated without build-up; consequently they tend to overestimate the effect of the region of reduced density, which in practice would be somewhat blurred by multiple scattering. A comparison of the density changes observed near an actual construction joint and the predicted increase in dose shows that to all intents and purposes such density changes can be ignored. In short, it appears as though the standards set for good quality civil engineering are already sufficiently stringent for reactor shield construction.

The design of any hole through the shield should, if possible, be such that the liner for the hole does not need to be positioned with very high accuracy. Although it is quite feasible in casting monolithic concrete shields to work to tolerances as close as 0.01 inch for the lining-up of experimental holes, to do so is expensive, because it calls for a rigid steel framework within the shield, or elaborate jigging, together with a great deal of supervision. A cheaper method, which resulted in considerable saving in the construction of the PLUTO reactor shield at Harwell, is to leave oversize holes in the concrete, through which the liners are inserted when the concrete is set. After positioning the liners correctly, the surrounding annulus is pressure-grouted with steel-shot concrete. With this technique one can rely on having no shrinkage voids between the concrete and the liner tube.

In designing the sections of buildings contiguous to the reactor shielding it is always advisable to avoid conditions of restraint because the concrete shielding is subject to continuous thermal movement once the reactor is in operation.

Research and Development

In view of the rapid developments in the design and actual construction of reactor shielding, no experimental verification has yet been carried out of the various proposals for their analysis. To make good this omission, various measuring devices have been incorporated in the concrete shield of the last reactor built at Calder and in various units of the first series of industrial nuclear power stations. These instruments include thermocouples, strain gauges, moisture gauges and crack width indicators to determine the temperature distribution through the shield, the strains in the concrete and reinforcement, the amount of moisture remaining in the concrete during operation and the rate at which cracks open at selected construction joints.

The installation of the instruments had been, at that time, part of an ad hoc programme to take advantage of particular circumstances and it was carried out by the Building Research Station in collaboration with the Atomic Energy Authority. Since then the programme has been considerably broadened to embrace more fundamental research, including laboratory studies. At Hinkley Point, the English Electric, Babcock & Wilcox, Taylor Woodrow Consortium is supplementing the site instrumentation of the reactor shielding with a comprehensive laboratory investigation of the site concrete. This covers most of its fundamental properties, such as strength, load strain, modulus of elasticity, drying shrinkage strain, moisture movement and thermal expansion, including hygrothermal shrinkage. Instrument readings on the site are being taken during construction and will be continued during starting up the reactor and its operation. It is hoped that this full-scale experimental work, supported by the numerical results of the laboratory investigations, will lead to a reasonable picture of the temperature regime, deformations of the structure, and the stress distribution, and thus facilitate a logical analysis and design of similar structures in the future.

One of the principal limitations of the work on the full-scale reactor described in the previous paragraph is imposed by the conditions of operation of the nuclear power stations. The behaviour of the shield at temperatures beyond those of normal operation can only be extrapolated from lower temparature values. Laboratory-scale experiments avoid this limitation, although scale work must inevitably introduce further complexity. Shrinkage, creep, and the time factor for temperature and moisture movements to reach equilibrium, for example, are functions of specimen dimension that cannot be truly simulated in scale-model work.

With the knowledge and experience available at present, model experiments can be carried out successfully to predict with reasonable accuracy the behaviour of structures under the usual external loads. Little has yet been achieved by model research probing into the influence of temperature and moisture. The instrumentation of several full-size reactor structures provides a unique opportunity of attempting the correlation of thermal and moisture behaviour of prototypes and their scale models. A programme of such model research has now been initiated and it is hoped that this work may lead to methods by which thermal and moisture behaviour of models may be interpreted in terms of the full-size structures.

In addition, the Building Research Station and various University Laboratories have embarked on a programme of fundamental research concerning Young's modulus and creep of concrete at high temperatures, and thermal properties of concrete for the various mixes and aggregates used.

References

B. T. Price, C. C. Horton, K. T. Spinney, "Radiation Shielding". Atomic Energy Research Establishment, Harwell. Pergamon Press. London 1957.

B. L. Goodlet, "Nuclear Reactor for Power Generation". The Institution of Mechanical Engineers. London, March 1956.

J. Bellier and M. Tourasse, "The Pressure Vessel of the Reactor at Marcoule". Bulletin d'Informations Scientifiques et Techniques. No. 20. Paris, August 1958.

A. C. Chamberlain and W. J. Megaw, "Safe Distances in Reactor Siting". A.E.R.E. HP/M. 109. Harwell, 1958.

G. Sege, "Containment-Vessel Design Basis for the Dresden Nuclear Power Station". Nuclear Engineering and Science Conference. Chicago, March 1958.

D. R. R. Dick, "Berkeley Power Station with Particular Reference to the Design of the Reactor Building". The Structural Engineer. London, March 1959.

Summary

The Paper gives a summary survey on the special problems involved in the design and construction of nuclear reactor structures, i. e., problems which are the concern of the civil and structural engineer. Siting of nuclear power stations, including their location underground, is discussed. The setting of the design basis for a reactor containment represents a novel task in engineering. Foundation problems are due to the extremely heavy weight of reactor buildings. Structural materials used in a reactor require special properties, such as resistance to radiation and corrosion. Design of pressure vessels, including prestressed concrete vessels, are discussed and the complex problem of radiation shielding is dealt with but very briefly. Certain engineering aspects, such as shrinkage cracks and uniformity of shield concrete, dimensional tolerances and structural restraints are considered. Finally, the current programme of research and the latest development work is described.

Résumé

L'auteur expose dans leur ensemble les problèmes particuliers qui se posent à l'ingénieur de travaux publics dans l'étude et la construction des centrales nucléaires. Il traite la question de leur implantation y compris la disposition en souterrain. L'établissement des bases du projet d'un ouvrage pour réacteurs constitue un aspect nouveau de l'activité de l'ingénieur. Les problèmes des fondations résultent des poids très élevés des centrales. Les matériaux employés doivent satisfaire à des exigences particulières, telles que la résistance au rayonnement radio-actif et à la corrosion. L'auteur étudie les cuves, en particulier en béton précontraint; il n'aborde que brièvement le problème complexe de la protection contre le rayonnement radio-actif. Il étudie certains aspects particuliers de la question, tels que la fissuration due au retrait, l'homogénéité du béton employé comme écran, les tolérances de dimensions et les restrictions constructives. Enfin, il donne des indications concernant les programmes de recherche en cours et les études les plus récentes.

Zusammenfassung

Die Arbeit gibt eine zusammenfassende Darstellung der speziellen Probleme, die sich bei Projektierung und Bau von Reaktor-Stationen für den Bauingenieur ergeben. Das Aussuchen des Bauplatzes unter Berücksichtigung der Anordnung als Untergrundstation wird diskutiert. Die Aufstellung von Projektierungsgrundlagen für ein Reaktorgebäude ergibt sich als eine neue Aufgabe des Ingenieurwesens. Die Fundationsprobleme sind eine Folge der

sehr hohen Gewichte der Stationen. An die Konstruktionsmaterialien für Reaktorgebäude müssen besondere Anforderungen gestellt werden wie Widerstand gegen radioaktive Strahlung und Korrosion. Der Entwurf von Druckgefäßen einschließlich solcher aus vorgespanntem Beton wird behandelt, während das komplexe Problem der Abschirmung der radioaktiven Strahlung nur kurz und unter Vorbehalt gestreift wird. Es werden gewisse Einzelheiten betrachtet wie Schwindrisse und Homogenität des Abschirmungsbetons, Abmessungstoleranzen und konstruktive Einschränkungen. Schließlich werden Angaben über das laufende Forschungsprogramm und die jüngsten Entwicklungsarbeiten gemacht.