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Pressure Vessel Codes: Their Application to Nuclear Reactor Systems

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1966

PRESSURE VESSEL CODES: THEIR APPLICATION TO NUCLEAR REACTOR SYSTEMS

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FINDINGS FROM A SURVEY BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY

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VIENNA, 1966

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FOREWORD

A survey has been made by the International Atomic Energy Agency of how the problems of applying national pressure vessel codes to nuclear reactor systems have been treated in those Member States that had pressurized reactors in operation or under construction at the beginning of 1963. Fifteen answers received to an official inquiry form the basis of this report, which also takes into account some recently published material. The report has been compiled by Mr. R. Skjöldebrand of the Division of Nuclear Power and Reactors of the International Atomic Energy Agency.

Although the answers to the inquiry in some cases date back to 1963 and also reflect the difficulty of describing local situations in answer to standard questions, it is hoped that the report will be of interest to reactor engineers, especially in those countries into which nuclear power is now to be introduced.

The Agency is grateful to the Governments of the Member States which contributed their experience to this report by answering the inquiry.

CONTENTS

I. INTRODUCTION

The design, construction and maintenance of nuclear reactor pressure systems pose a number of new problems in connection with the application of normal pressure vessel codes and regulations. In part, these problems concern new aspects of pressure vessel techniques such as the very large sizes of the presently produced and designed reactor vessels, the high pressures and large volumes, and thus the large stored energies in the systems. Often unconventional materials and fabrication techniques are used, e.g. materials such as Zircaloy, prestressed concrete or multi-layered vessel walls. The main problems, however, are: the vessels undergo service in high radiation fields, which affect the materials and cause their activation; they are inaccessible for routine inspections once they have been put into operation; they contain very large amounts of radioactive materials in operation; and there can be very grave consequences in the event of leakage or failure. Finally, nuclear reactors are often sold internationally, and it can sometimes be very difficult for a manufacturer to understand and meet the requirements of foreign codes. Typical of these latter problems is a statement made at the Third Geneva Conference by a Dutch manufacturing group [1]: "Not all countries have rules for nuclear vessels, thus lengthy discussions with the authorities concerned are required to establish a set of rules. In the second place, where one has to design a vessel to more than one code, conflicting requirements have to be discussed between the authorities concerned. The conventional codes in most cases do not give sufficient information on the design requirements of nuclear vessels, so additional, provisional rules are given or calculation methods have to be agreed upon and much detailed analysis is asked for".

Pressure vessel codes can indeed be of crucial importance for the design and construction of nuclear reactor vessels. For the sizes and design pressures now considered for big power stations in the 1000-MW range, a national code interpretation or ruling may not only severely affect the cost of the plant but also in fact make it technically unfeasible to build and inspect a certain reactor in one country while it may be possible in another. Figure 1 shows the size ranges for built and planned boiling and pressurized light-water power reactor vessels.

As is shown by recent developments, these big-size vessels are necessary to achieve economic competitiveness for nuclear power. The data in Table I are illustrative for the wide variations that are obtained in a simple calculation of the wall thickness as required by various national codes for typical modern power reactor vessels. The BWR in this table reflects design data for the Oyster Creek plant and the PWR the Connecticut Yankee station, both now under construction.

The problems of applying pressure vessel codes to nuclear reactor systems will be encountered much more often and on a larger scale in the future. It is expected that by 1980 the installed nuclear generating capacity will be more than 200 000 MW(e), which would correspond to at least some 200 to 300 new nuclear power plants being in operation by that time. Fifty-nine of them are already firmly planned or under construction. Many of these reactors will be in developing countries which do not have the level

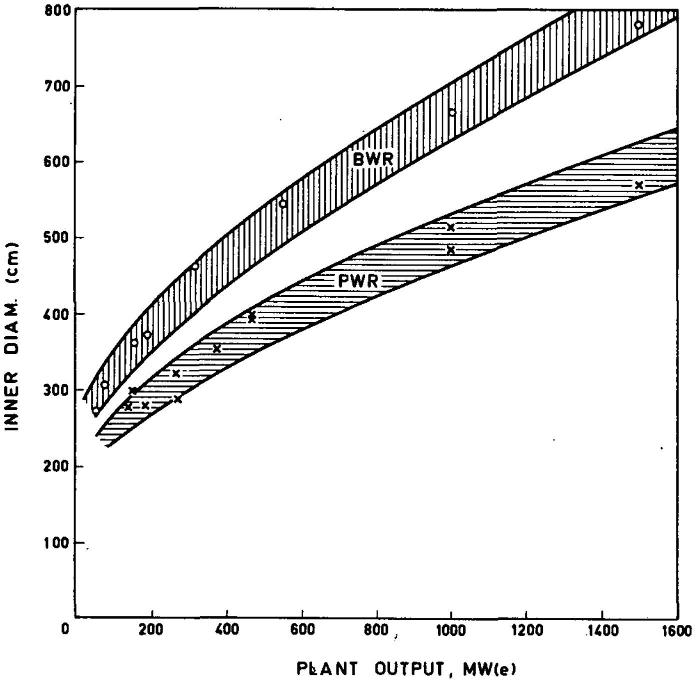

FIG. 1. Diameter of light-water reactor vessels

of general technological development that has eased the problems of pressure vessel design and construction in, e.g., the United Kingdom or the United States of America.

At the same time the very rapid development in nuclear reactor technology will make it difficult to adapt existing national codes through supplements or case rulings for nuclear reactor systems or to formulate new codes to cover reactor system applications in a general way. New codes should cover the widely different cases of water-moderated and cooled reactors on the one side and gas-cooled, graphite-moderated reactors on the other, all of which systems have been and are being built for nuclear power stations. All these systems are in a stage of rapid development, e.g. towards internal nuclear superheating of the steam in water-cooled systems and towards very high exit temperatures of more than 700°C of gas coolants. The coolant parameters of the present advanced reactor concepts are such that new problems of design and manufacture are raised also for the more conventional parts of the reactor station systems. All the same, the present generation of reactors will most probably be superseded within some decades by entirely different types, such as advanced converters or sodium-cooled fast breeders, which will place new and still more different requirements on the pressure vessel systems.

The problems posed by the application of the many different national codes have already in the past been severe and it was suggested that the International Atomic Energy Agency make a survey of how pressure vessel

TABLE I

SCHEMATIC CODE EXAMPLES FOR TYPICAL LIGHT-WATER REACTORS
BWR, Oyster Creek Plant (515 MW(e))
PWR, Connecticut Yankee Plant (462 MW(e))

	BWR	PWR
Inner diameter (cm)	541	391
Design pressure (kg/cm^2)	88	176
Material, ASME spec.	SA-302-B	SA-302-B
Coolant outlet temperature (°C)	285	308
Wall thickness (mm):		
(a) As actually built	181	270
(b) Designed to code of		
Australia	254	384
Belgium	242	352
Denmark	274	396
Germany	173	270
Norway	226	326
Sweden	194	336
USSR	210	303
USA (ASME VIII)	175	264
INSTA	194	-

codes have been applied in those Member States with experience in the construction of pressurized reactor systems. An inquiry was sent out and by the middle of 1964 fifteen answers had been received. Some additional background literature has been reviewed and included, but the report is mainly based on these answers, which means that the information in some cases dates back to 1963. Further, the answers were very different in the scope they covered, and in the amount of detail they gave. In many cases the specific technical examples have been taken from the literature to support the more general considerations referred to in the official answers to the Agency's inquiry. Both water-cooled and moderated and gas-cooled reactors are covered by the answers, i.e. both water pressure vessels or boilers and gas pressure vessels. The answers naturally also reflect local situations and the difficulty of describing them in reply to a standard inquiry. The report thus cannot claim complete coverage. Nevertheless it is believed that the material can give some background for the work on code problems now going on in many countries and that it may be of some assistance in the countries that are now planning the installation of their first power reactors.

II. LAWS, REGULATIONS AND REGULATORY STANDING OF CODES

In most countries the design, construction and inspection of pressure vessels in general are subject to government regulations, but the type of regulatory control varies within very wide limits. This will sometimes have a bearing on how the regulations and codes are applied to reactor systems, and the pertinent facts for each of the countries which answered the Agency's inquiry have therefore been collected in the Annex.

To illustrate the very different positions in various countries, just some examples can be given here. In Belgium a royal decree constitutes the whole code and in India and Japan the codes are ministry regulations, written and issued by government departments. On the other hand, in the United Kingdom and in France, the laws give only a minimum of guidance on technical questions, placing a general responsibility for safety on the owner/operator of the plant. Most countries take a position in between these extremes with a law establishing principles only, referring detailed problems of design and construction to codes written by a technical organization. Thus in Norway the law establishes three mandatory codes, the Dutch law gives the code regulatory standing, while in Denmark and Sweden the codes have no legal standing per se but are in practice official and mandatory as they have been adopted by the government departments that are by law responsible for labour protection. In federal unions such as Australia, Canada, Germany and the United States of America the regulatory standing of the codes is determined by the separate states or provinces in the union, where most often the procedure of a state law establishing a technical code has been followed.

In a few countries, express provision is made for the acceptance of foreign codes. Thus the Australian Code CB1/1962 Part II, dealing with unfired pressure vessels, recognizes some British and US national specifications and codes as acceptable, among them British Standards, Lloyd's Rules and Regulations, and ASME[1] Codes. The Canadian Code B51 specifies the use of ASME Codes with some amendments.

The administration and enforcement of the codes likewise vary greatly. Most countries have inspection organizations, which may be part of a government or state department (Australia, Canada, Denmark, France, India, Japan, Netherlands and Norway), or may be separate technical organizations recognized by a government or state department (Germany and Sweden). In Belgium recurring inspections are performed by authorized private organizations. In the United States of America initial inspection is performed by state inspectorates or by certified inspectors. In the United Kingdom the general enforcement of the law is by the Ministry of Labour (for nuclear installations by the Ministry of Power) but detailed inspections are made either by the purchaser, by insurance inspectors, or by other independent technical inspection bodies.

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[1] ASME = American Society of Mechanical Engineers.

In some countries, e.g. Canada, an insurance would annul the requirement for periodic inspections by authorities, but the insurance companies in such cases would perform similar inspections.

III. SCOPE AND CONTENT OF NATIONAL CODES

In Table II the scope and content of the national regulations and codes have been summarized as far as is possible with the information that is available. To provide comparative material, both regulation and code scopes have been combined in the table without any attempt at separation. It is obvious from Table II that great differences exist between the countries and that this would cause problems for international deliveries even for conventional vessels that do not have the advanced requirements of nuclear vessels.

For nuclear applications it can, for instance, be seen that many codes do not have material specifications for the modern low-alloy, high-strength steels, which are the ones that must be used for big reactor vessels. Further, many codes do not give any design stress values even for the temperature ranges of present water-cooled reactors, and very few give values in the higher temperature range of modern gas-cooled reactors, for which creep is becoming important. The given stress values at elevated temperatures vary considerably. Few codes give specifications for aluminium and its alloys, which are extensively used in research reactor systems.

Most of the codes give extensive coverage to welded vessels, welder qualification, manufacture by welding and inspection of welds. The forms of welding control vary very much, ranging from control of the welder's capability to specifications for non-destructive testing. Requirements of test plates to be welded with longitudinal welds are quite common for quality control. Radiography is the only non-destructive testing technique which is commonly recognized, ultrasonic techniques being mentioned in some codes without any detailed specifications.

An excellent survey of national pressure vessel codes, giving more detailed information, has been published by Lancaster [2].

IV. THE APPLICATION OF NATIONAL PRESSURE VESSEL CODES TO NUCLEAR SYSTEMS

All the 15 Member States that answered the IAEA enquête stated that the national pressure vessel regulations and codes had been applied to the nuclear pressure vessel systems built in these countries.

Nineteen of the reactors concerned are imported into nine countries. In some of these cases the application of the codes was particularly easy, as provisions existed or were made for the acceptance of foreign codes. Australia is a particularly clear-cut case, as the Australian SAA[2] Code makes express provisions for the acceptance of the British Standard Specifications to which the HIFAR was designed and built in the United Kingdom.

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[2] SAA = Standards Association of Australia.

TABLE II

SCOPE AND CONTENTS OF NATIONAL PRESSURE VESSEL CODES AND REGULATIONS

	Materials specifications						Code design basis					Welding		Inspection procedures		
	Carbon steel	Low alloy *steel*	High alloy *steel*	Non-ferrous metals	Values for high temperatures	Tests and inspection of *materials*	Formula for minimum wall thickness	Complete stress analysis required	Allowable stress based on		Joint efficiency factor range	Welding approval *procedure*	Tests of *welds*	First inspection	Periodic inspection	
									Yield strength	*Ultimate* strength						
Australia	Yes	No	Yes (Cu)	Yes	Yes	Yes	Yes	No		Yes	0.41-1.0	Yes	Yes	Yes	Yes	
Belgium	No	No	No	No	No	Yes	Yes	No		Yes	< 0.7	No	No	Yes	Yes	
Canada																ASME Code applies welding requirements slightly modified in B 51
Denmark	Yes	No	No	No	No	No	Yes	No		Yes	0.5-0.9	Yes	Yes	Yes	No	
France	Yes	Yes	Yes	Yes	Yes	-	Yes	No		Yes	-	Yes	Yes	Yes	Yes	
Germany	Yes	Yes	Yes	No	Yes	Yes	Yes	No	Yes		0.8-1.0	Yes	Yes	Yes	Yes	
India	-	-	-	-	-	-	-	-	-	-	-	-	-	Yes	Yes	
INSTA	Yes	No	No	No	No	Yes	Yes	No	Yes		0.4-0.9	Yes	Yes	Yes	No	
Italy	Yes	Yes	Yes	Yes	-	-	Yes	No	Yes[a]	Yes	-	-	-	-	-	
Japan	-	-	-	-	-	-	-	-	-	-	- -	-	Yes	Yes	Yes	
Netherlands	Yes	Yes	No	No	Yes	-	Yes	No	Yes	-	-	Yes	Yes	Yes	-	
Norway	Yes	No	No	Yes	No	Yes	Yes	No		Yes	0.5-0.9	Yes	Yes	Yes	Yes	
Sweden	Yes	Yes	Yes	No	Yes	No	Yes	No	Yes		0.6-0.9	Yes	Yes	Yes	Yes	
UK	-	-	-	No	Yes	-	Yes	No		-	-	-	-	-	-	
British Stand. 3915	Yes	Yes	No	No	Yes	No	Yes	Yes	Yes	Yes	0.95-1.0	Yes	Yes	Yes	No	
USA ASME II, VIII and IX	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	0.45-1.0	Yes	Yes	Yes	No	
ASME III	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	1.0	Yes	Yes	Yes	No	
USSR	Yes	Yes	Yes	Yes	Yes	No	Yes	No	Yes		0.7-1.0	Yes	Yes	Yes	No	

- *Information not available.*

[a] The proposed new code will use yield strength.

In Denmark British and American codes were accepted for the reactors in operation at the Risø Research Establishment and in Greenland, although the Danish Atomic Energy Commission made some special additional requirements. In Germany, where in general the national codes have been applied, foreign codes have been accepted in some specific sectors, e.g. regarding special designs.

Some quite severe difficulties of a general nature have in many cases been encountered in applying national codes to reactor systems. This in many cases stems from the fact that the local codes are old and do not reflect the technical development that has taken place in the design and construction of big pressure vessels. Sometimes this results in the national codes being too strict, e.g. in not recognizing the high strength of modern pressure vessel steels, and sometimes they are entirely inadequate in that they do not cover modern types of construction of high pressure, high power units. The mere sizes and wall thicknesses of reactor vessels in many cases go far beyond existing practices and have brought new requirements beyond those of existing codes. It should also be noted that sometimes codes which are not truly appropriate have to be applied. In India, for example, no code for unfired pressure vessels, under which reactor vessels and systems are normally treated, has existed. The Tarapur BWR plant should, as a steam generating unit, be designed and manufactured according to the rules of the Indian Boiler Regulations, which in many aspects would not be applicable, as the reactor plant naturally is very different from any conventional boiler.

From the United States of America it was pointed out that one major problem in applying stringent codes to nuclear vessels has been that of obtaining close adherence of all manufacturers and vessel inspectors to the requirements of the codes in order to avoid delays when such requirements are enforced. Similar difficulties have indeed been experienced in many countries, as delays and rejections in nuclear plant deliveries bear witness.

In addition to these more general problems which would be similar for both big conventional units and nuclear vessels, there are some major ones which derive from the special characteristics of nuclear reactors. The competence to deal with these problems has normally been developed within national atomic energy authorities, which have been charged with the national nuclear power development programmes. Often these authorities have also been given a safety control function, which in others is handled by special atomic energy control boards or reactor safety committees. A reactor safety committee may sometimes be only advisory, e.g. to a government department or an atomic energy authority organization. In other instances the exercise of the safety control function by a control board or directly by a government department will have a definite regulatory character. Pressure vessel safety is usually considered as part of nuclear safety under this control function and it is thus interesting to study how this is correlated to the administration of pressure vessel codes. Procedures and practices vary very much, as can be seen from the following examples.

In Canada the regulatory control over all nuclear reactors is the responsibility of the Atomic Energy Control Board, which licenses a reactor only after a thorough review of its safety. This Board seeks the advice of appropriate provincial authorities regarding boilers and pressure vessels.

Japan applies its normal codes to research reactors but not to power reactors, which are instead subject to approval of the Ministry of Trade and Industry, in accordance with a special regulation for power reactor facilities.

It is interesting to note that in some countries an act may prescribe nuclear licensing procedures that tend to give codes a regulatory effect they do not otherwise have. A case in point is the Nuclear Installations Act of 1965 in the United Kingdom.

Naturally, intervention by nuclear energy authorities or control bodies to deviate from usual code practices in nuclear plant licensing has occurred to the greatest extent in countries where the problems of applying the national code have been most severe. One example is India, where the Atomic Energy Act provides for the possibility of superseding the Boiler Regulations in relation to any facilities owned by the Central Government and engaged in carrying out the purposes of the Atomic Energy Act. Difficulties were encountered with the application of the Indian Boiler Regulations to the Tarapur plant which is being imported from the United States of America. Several exemptions to these regulations would have to be made concerning material specifications, weld factor, general design, etc., and it is possible that an ASME code will be adopted for this plant instead. It is thought possible that with this background an independent inspectorate will be established under the Indian Department of Atomic Energy, and also that an appropriate code will be drafted for the guidance of this inspectorate. It is quite probable that this is a solution to the problem of old or inadequate codes, to which recourse will be taken in many countries as the first power reactors are built.

V. NEW NUCLEAR CODES AND CODE SUPPLEMENTS

In many countries the pressure vessel code writing bodies recognize the need to update or re-write the codes so that they specifically cover nuclear pressure systems. In the United States of America the ASME Boiler and Pressure Vessel Codes were supplemented by special nuclear reactor case interpretations [3], the earliest of which dates back to 1959. In 1964 a new Section III "Nuclear Vessels" of the ASME code was published. The "Provisional Requirements for the Design, Manufacture and Testing of Land-based Nuclear Installations", published by Lloyd's Register of Shipping 1960 [4], has been used in the United Kingdom, where the British Standards Institution published B.S. 3915 "Specification for Carbon and Low Alloy Steel Pressure Vessels for Primary Circuits of Nuclear Reactors" in 1965. In Germany it is planned to amend the steam boiler and pressure vessel codes by supplements to the "AD Merkblätter". A series of amendments to the existing codes to cover nuclear cases are also planned in Sweden.

The ASME Section III, Nuclear Vessels, Code [5] merits special attention. It has been discussed more fully by, among others, Gaines and Porse [6] and thus only some of the major features will be pointed out here to serve for comparison with some of the requirements raised for nuclear vessels in other countries. The ASME Section III Code introduces a new concept of design compared to the earlier Sections I and VIII. The design criteria are different in the following respects [7]:

"(a) Section III uses the maximum shear stress (Tresca) theory of failure instead of the maximum stress theory.
(b) Section III requires the detailed calculation and classification of all stresses and the application of different stress limits to different classes of stress, whereas Sections I and VIII give formulas for minimum allowable wall thickness.
(c) Section III requires the calculation of thermal stresses and gives allowable values for them.
(d) Section III considers the possibility of fatigue failure and gives rules for its prevention."

Because of the complete stress analysis that it requires, Section III allows lower safety factors. Vessels designed to this code thus show savings in material compared to those designed to the older code sections. Material savings of 16% for a PWR reactor vessel and 25% for a BWR reactor vessel have been quoted [6].

Section III also requires more extensive testing of the materials used, and recommends that the increase in the brittle fracture transition temperature due to radiation damage, be checked periodically by means of surveillance specimens irradiated in the reactor, also considering combined effects of fabrication, stress, and neutron exposure. Requirements on fabrication and inspection are finally more restrictive in Section III than in the earlier code sections, reflecting the practices that have been developed in the American industry for the manufacture of nuclear vessels.

The new ASME Section III Code thus aims at a thorough analysis of the designs and knowledge of the completed vessels, giving instead considerable savings in material and fabrication. This trend to replace large factors of safety with careful and sound engineering is significant. The same concept is, for example, the basis for the new British Standard, and for a new code supplement for nuclear cases being drafted in the Netherlands which will have even higher design stress values than ASME Section III.

VI. MAJOR CONSIDERATIONS FOR NUCLEAR PRESSURE VESSELS

Some major general considerations concerning the special characteristics of reactor systems have been the basis for special requirements for nuclear vessels in most of the countries concerned. They are as follows:

(a) Fast neutron irradiation will cause an increase in the brittle fracture transition temperature in steel. The exposure range needed for this effect to become significant is one which can be reached in the walls of a nuclear reactor pressure vessel during its projected lifetime.
(b) Owing to radioactivity, contained in both fuel and coolant and induced in vessel walls, it is often impossible or extremely difficult to perform the periodic routine (e.g. annual) inspections that are required by most pressure vessel regulations or codes.

The effective containment of large amounts of radioactive material is one major concern specified by, among others, Denmark, Germany, Japan and the United States of America, where nuclear vessels previously were classified as "vessels containing lethal substances" for design, construction and inspection according to the ASME Section VIII Code. The same major consideration is also reflected by Lloyd's "Provisional Requirements" [4] and [8].

In addition to these considerations, which applied to most of the answers received, thermal stresses in the walls due to changing operating conditions (Germany and USSR [9]) and to radiation heating of the wall material (Germany, Sweden and USSR [9]) have been reported as a major concern.

As a result of these considerations, both the new ASME Section III Code and Lloyd's "Provisional Requirements" recognize the need to distinguish between different classes of vessels in a nuclear plant system. Lloyd's thus defines Category A vessels as those that contain radioactive material, are subjected to high-level neutron irradiation and are inaccessible for periodic maintenance and inspection. Category B are vessels which contain radioactive materials in service but can be made available for periodic maintenance and inspection during shut-downs. Category C covers the parts of the system that are to function as normal severe-duty pressure vessels (Lloyd's normal Class I). Emergency steel containment vessels fall under Category D. The ASME Section III, Class A, would correspond roughly to Lloyd's Category A and B, Class B to Category D, and Class C to Category C. B.S. 3915 similarly distinguishes between two categories of vessels in nuclear systems.

In the answers received to the Agency inquiry, no distinction has generally been made between any such categories of nuclear system components. In the text of this report, the only distinction that is made is thus between the reactor vessel itself and the nuclear pressure vessel systems, which would correspond to ASME Class A.

Special requirements due to radiation embrittlement of steels

The general considerations above also lead to some general requirements quoted in many of the answers to the IAEA inquiry. The process of radiation embrittlement of steel influences the choice of material, the allowable operating conditions of the vessel and its operational lifetime. Major testing programmes of irradiated test specimens have been performed to ensure that the pressure vessel steels used are well known experimentally (e.g. Italy, Netherlands, Sweden, USA [6]).

In many countries irradiation of reactor vessel steel specimens inside the reactor during its operational lifetime is also required (e.g. in Germany, Italy, Sweden, UK) or recommended (by the USA in ASME Section III). Very extensive irradiation programmes for investigations on surveillance specimens of the reactor vessel material have been reported by the United Kingdom and the United States of America [10] where the tests include corrosion, impact and tensile strength on samples from the vessel plate as well as weld metal and forged materials. Samples are removed and tested at planned intervals during the whole lifetime of the reactor vessel. This, however, also brings out the difficulty in interpreting irradiation test results. Embrittlement is not only a result of fast neutron irradiation but also of normal ageing, strain ageing, influence of heat-affected zones in the neighbourhood of welds, etc., which may cause equally large or larger changes in the transition temperature and about which, of course, irradiation test specimens give no information. Further, for radiation-induced embrittlement previous heat treatment of the material, the impinging neutron spectrum and the irradiation temperature are important parameters that should be the same for the test specimen as for the vessel wall material about which they are to give information. Dose-rate effects have also been reported but results are not consistent. The normal and miniature Charpy V-notch specimen tests, normally used in such applications, make it possible to compare the irradiation behaviour of different steels but do not give enough information for a thorough evaluation on an absolute scale of, for example, the actual behaviour of a reactor vessel under irradiation, the suitability of a certain steel, the lifetime that can be safely estimated for a vessel, or minimum safe working temperatures.

The question of how to evaluate the suitability of vessel steels to meet reactor vessel operating conditions has been approached in different ways. In Sweden, for example, it is suggested that the isothermal Robertson crack-arrest temperature test be used as a basis; that is, if the crack arrest temperature for irradiated specimens is found to be very much lower than the lowest temperature at which the vessel will be pressurized, then there is a large safety margin and the steel is adequate [11].

Also the problems of lifetime and operational limitations have been approached in several different ways and so far no clear common practices have been developed. Normally, shields are used to reduce vessel-wall exposure; in Japan, for example, such shields are mandatory. In Norway the Halden reactor vessel has been estimated to be safe to a total fast neutron exposure of 10^{18} nvt. In the United States of America a survey has been made of the fast neutron exposure of some power reactor vessels [12]. Of the 19 reactors surveyed, 15 may experience exposures of more than 10^{18} nvt during their lifetime, and 8 may receive exposures of more than 10^{19} nvt. The PM-2A reactor was the first to reach 10^{19} nvt; it has now been taken out of operation.

Another approach would be to design the vessel to the worst possible conditions. The first Swedish power reactor, at Ågesta, was designed at an early period, when it was felt that not enough was known about radiation damage effects on the material used and it was thus decided to assume the whole vessel brittle, that is, all calculated stresses are below yield [13, 14].

In the United Kingdom it has been the practice to reduce the risk of brittle failure by maintaining the reactor pressure vessels at a certain minimum temperature, which is assessed from crack arrest tests, changes in this temperature being estimated from Charpy specimen transition temperature changes [15]. Operational criteria, based on a permissible area in a pressure-temperature diagram have also been used in the United States of America [12] and similar considerations have been reported from Germany [16] and Sweden. It would seem desirable to investigate further the needs for such operating criteria and practices which are in use at present.

It has also been proposed to include other new destructive tests on the materials to be used for the vessel, such as determination of fracture toughness properties as function of neutron exposure [6].

An alternative to measurements on test specimens, irradiated inside the vessel, could be to perform measurements on the vessel itself as it is pressurized, as its temperature is increased, etc. At the Kahl BWR in Germany an experimental programme includes elongation measurements on the internal and external surfaces of the pressure vessel during operation.

Requirements due to inaccessibility for periodic inspections

The difficulty or impossibility of performing regular periodic inspections on reactor systems after they have been taken into operation has led to a great number of additional requirements beyond those given in national codes. While radiation embrittlement, being a change in the property of material, will mainly have an effect on material considerations, the inaccessibility for inspection demands additional requirements for the whole fabrication from the design to the final acceptance tests.

Many of the answers to the IAEA inquiry stated that regular inspections should continue to be performed as far as possible. The United Kingdom answer points out that it is not possible to give any general rules for regular inspections, but that regulations may be limited to require the observance of any examination conditions laid down by the licensing authority. This, of course, means a "case-by-case approach" which is, however, typical of most nuclear plant licensing procedures and which reflects the present state of development in the nuclear technology.

Typical of the inspection rules thus established are the Australian requirements on the HIFAR research reactor system, which include repetition of the acceptance tests for components when they are removed or dismantled for inspection and maintenance. Regular visual inspections of the reactor vessel are performed during each annual major maintenance shut-down. Similar requirements are in force in other countries.

It may sometimes be possible to isolate nuclear system components, such as steam generators, and to perform a normal periodic inspection on them. In most cases, and especially as regards the reactor vessel itself, periodic inspections would have to be very limited in scope. It is often only possible to perform a superficial visual inspection during major annual maintenance shut-downs, using boroscopes or other optical aids; this is much less than most rules for periodic inspections specify.

Regular inspections, as normally conducted on conventional boilers, being impossible, most countries have naturally required higher standards of reliability for reactor vessels. These higher standards in turn call for a more extensive knowledge of the static and operational properties of the vessel, i.e. a careful stress analysis and a thorough evaluation of all effects that may be deleterious. Thus, corrosion resistance and analysis of all phenomena that may endanger the structure in the long term have generally been required.

Most countries have required a careful analysis of all stresses in the vessel beyond the simple calculation of wall thickness specified in the codes.

The objective of the stress analysis has been different in various countries. Sometimes the purpose has been to perform calculations in order to discover and ensure against inadequacies in existing codes for the more complex nuclear reactor vessels (Italy), or to determine the real factor of safety, also taking into account radiation damage (Norway). The Norwegian answer also quoted that as a result of these investigations it was decided to use a maximum weld efficiency factor of 0.75 longitudinal welds in the design calculations.

Other answers state that the usual schematic code calculations, taking a limited value for membrane stresses with a safety factor, leaving all uncalculated stress increases to be carried by the margins available, have been considered unsatisfactory for reactor vessels, which require a detailed analysis of all stress increases due to openings, changes in temperature, etc. (Germany, Netherlands, Sweden [13], UK). Thermal stresses have in some cases been given special attention (e.g. Germany, USSR [9], USA). The Japanese answer specifies that thermal stresses have to be considered in the maximum allowable working pressure, and the German answer points specifically to the additional thermal stresses that may occur in thick walls at steady-state operation and especially during power changes in the reactor, as well as to the occurrence of additional thermal stresses due to radiation heating of the materials. This has also been a main consideration in the USSR [9].

For the Marviken reactor vessel in Sweden a complete stress analysis had to be made, taking into account all normal reactor operating conditions (including fast shut-downs) and also some types of accidents. For these it was required that plastic deformation of the components would not be so great that their function would be impaired. This is the only reported instance of special requirements for reactor accident conditions.

Different types of reactors give somewhat different requirements as concerns the stress calculations. Creep is not covered by the ASME Section III Code except in the way that the tabulated allowable stress intensities have been limited to a temperature range below that at which creep behaviour becomes dominant. This range is still adequate for the water-cooled reactors now forming the main part of the United States commercial power reactor production. In the United Kingdom where gas-cooled reactors call for higher coolant and vessel temperatures, creep is a major concern. Similarly, some concern about the adequacy of the earlier ASME code sections has been reported from the United States of America for high temperature liquid-metal cooled systems [17].

It has been stated that a complete stress analysis of a vessel design may add up to 20% to the cost of the vessel [18].

Another significant action taken to counteract the inaccessibility for regular inspections is the research programme that has been established in the United Kingdom by the Central Electricity Generating Board to predict the performance of steel reactor pressure system components [19]. The main areas of interest in this investigation are creep distortion, creep rupture, and high strain fatigue. Some experiments have also been made to investigate the margins between actual operating conditions and those which would create distortion of the components by short-term yielding.

Additional requirements have also been placed on the whole manufacturing procedure, from the selection of materials to the stress relieving procedure, and on the inspection. Of course various inspection procedures are often emphasized, as the objective behind most of these requirements is the verification that the strict specifications have been maintained in construction. This has often meant the development of new or more advanced non-destructive inspection techniques, e.g. the use of accelerators for radiography, ultrasonic methods (which are now specified to a much greater extent than previously), new leak tests, etc. These new or more refined techniques are now being further applied to the fabrication of more conventional pressure vessels.

For all steps in design, fabrication and inspection there is also a general call for careful and extensive documentation of all actions. The various requirements are sometimes typical for specific reactor types and may not have general validity.

Among the requirements quoted in many answers are:

(a) Special care must be exerted in the specification, selection and inspection of materials to be used, to avoid laminations, inclusions and other defects. Inspection, both as regards chemical analysis, destructive and non-destructive testing (e.g. by ultrasonic methods) is often stricter than usual and well specified (e.g. Germany, Japan, Sweden, UK, USSR [9]).

(b) For welded constructions, special requirements have been set up for welder qualification, preheating, and weld inspection (e.g. UK). Full penetration welds are normally specified. In Canada it has been specified that the welds must have a configuration such as to permit radiographic examination. One hundred per cent radiography of all welds is normally specified (e.g. Belgium, Germany, Japan).

(c) The stress relieving operation is to be checked, and strain, temperature and vessel movement records are to be taken (UK). Additional stress relief requirements are sometimes laid down (Canada).

(d) The hydraulic tests are in many cases performed more strictly than normally and under strict supervision (Belgium, UK). It has been pointed out that hydraulic testing to excessive pressures may be undesirable and result in undue strain on the vessel (Australia).

(e) Emergency reactor containment vessels are often specified and built as pressure vessels. The pressure tests on these, however, are most often adapted to their special function, which is to serve as a barrier against leakage in adverse conditions, rather than to safely withstand a certain pressure. These vessels are thus usually periodically leak-tested at design pressures, rather than tested at a certain over-pressure.

(f) Leak tightness has been given special attention in many cases (Australia, Denmark, Germany, Japan, UK). The degree of leak-tightness required is not always determined by safety in containment of radioactive materials, but also by the possibility of loss of expensive coolants, such as heavy water and helium. Leak tests thus vary very much from soap water tests through vacuum tests to freon and helium-gas leak tests.

It is interesting to note that original strict requirements now in some instances are being relaxed, as experience adds knowledge and shows such relaxations to be permissible. One example of this is the early requirement for corrosion resistance in the primary systems of reactors. In some reactors unclad carbon steels are now used since extensive experiments show that corrosion rates can be controlled and do not introduce unknown risks during the reactor lifetimes (USA and USSR). The background for such relaxations is, however, always very extensive knowledge.

Additional requirements

A number of more specialized requirements have also been given by some countries:

Canada:	Additional ultrasonic inspection of welded joints was required. Castings and repairs to castings were subject to additional radiographic, ultrasonic and dye-penetrant inspection.
Germany:	A welding factor of 1.0 was aimed at by adopting the code's strictest welding and design requirements. Special specifications were in many cases given for surface finishes.
Japan:	Special attention is given to safety valves, and to the safe disposal of radioactive wastes from them.
UK:	A thorough visual examination of the vessel after strain relief is required, followed by magnetic crack detection of suspected areas.
USSR: [9]	Ring-shaped portions of the vessels are forged without any longitudinal joints. All nozzles are double-welded to the vessel.

Exceptions granted from existing codes

Although the national codes thus have been generally applied, some notable exceptions have been made. With the French G-2 and G-3 reactors it was obviously necessary to allow a great latitude in the code interpretation as they have prestressed concrete reactor pressure vessels of an entirely new design concept.

The French answer also stated that, although the "Service des Mines" normally is the control body for pressure vessels, including nuclear systems, liquid-metal cooled fast reactor systems have been exempted from that control in view of the very special nature of these vessels.

Many smaller exceptions were reported, pertaining to only certain aspects of the design or construction as, e.g., for the French EDF-1 and EDF-2 reactors, for the Indian Tarapur station, and for the Norwegian Halden reactor primary system.

VII. CONCLUSION

It has been possible to apply the normal national pressure vessel codes to most of the pressurized nuclear reactors built so far, and the bodies normally responsible for the administration of codes and regulations have still had this function to fulfil, even if sometimes the procedures differ from the routine for boilers and conventional vessels.

Code applications to nuclear systems have caused severe problems. Generally many additional or separate requirements have been made beyond those of the codes. Even these countries that have modern and technically well-advanced codes have had to stretch the requirements and extend the scope of these codes.

It is, of course, only natural that many of the existing codes seem old-fashioned when they are applied to a field in which such rapid technological development is taking place as in nuclear power engineering and in which new problems are raised by the special operating conditions. Nuclear vessel applications have also highlighted shortcomings of the present codes, above all the inadequacy for big and complex vessels of the schematic design approach typical for almost all codes.

The criticism of this simple approach, which prescribes calculation of wall thicknesses of certain given vessel shapes by a simple formula, is based on its over-simplification of actual conditions. It takes into account a too limited number of physical material properties. The safety factor is large, often resulting in unnecessarily thick walls with accompanying fabrication difficulties and costs, because it must cover a wide variety of contingencies such as design aspects, lack of full inspection of the vessel and knowledge of the design and, finally, quite often simply bad design. The complex cases of nuclear vessels have brought out the insufficiencies of this approach and have required a complete stress analysis of the vessels. Only in a few cases, however, has the consequence been drawn that the complete stress analysis, representing good design engineering against specific possible causes of failure, should permit more realistic safety factors to be used. As a general rule, the requirements of the old codes have still been the minimum ones.

The reluctance to deviate from old codes can easily be explained by the good experience of accident prevention that they have given. It has also recently been pointed out that we are not yet ready with our present techniques to discard completely the old "design stress" concept to design against failure of any kind starting from basic material properties [20]. Still, the insufficiency of the old approach has been clearly realized in nuclear cases as well as for more conventional vessels [21] and important steps have been taken towards more qualified design rules in the ASME Section III Code, in the work by the ISO working group now occupied with formulating a new code for nuclear vessels, and in several new national nuclear codes and code supplements.

The additional requirements beyond those of the existing codes, which have been made for nuclear vessels, have, to a great extent, involved development of more careful design techniques, new and improved fabrication methods and new inspection procedures. Good engineering and quality control have been very much emphasized in the construction of nuclear vessels. This has, beyond any doubt, benefited the whole field of pressure vessel construction and may well be reflected in amended or new codes for non-nuclear vessels.

Important aspects of reactor vessel operation, such as lifetime and limitations in operating conditions, although they are closely related to code problems, cannot be covered in codes as they are dependent on the varying operating conditions in each reactor. These aspects must, however, be given serious consideration and a more unified approach to them would seem very desirable.

Many countries are now devoting efforts to formulating new codes or code supplements to cover nuclear vessel cases. These may show a natural tendency to reflect the locally chosen line of development for reactor systems. With the variety of very different systems that already exist and which will become greater as work continues on advanced converters and breeders it must be hoped that enough flexibility in code formulation is retained to permit up-dating and application to these systems.

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ANNEX

NATIONAL REGULATIONS AND CODES AND THEIR ADMINISTRATION

AUSTRALIA

The principal code is the Australian Standard Rules for the Design, Construction, Inspection and Operation of Boilers and Unfired Pressure Vessels and their Appurtenances, Australian Standard No. CB. 1, published by the Standards Association of Australia. It is commonly known as the SAA Boiler Code.

This code has no legal or regulatory status by itself, but it has been written into the statutory requirements of some of the Australian States. In all cases boilers must be registered and have a certificate of worthiness, which is to be renewed at periodic inspections.

New South Wales, Queensland, Victoria, South Australia and Tasmania thus have acts which give the SAA Boiler Code legal status, in some cases giving others, such as relevant British Standards, Lloyd's rules or ASME code as alternatives. Where the code as yet has not been given this legal standing the inspection authorities have developed a policy which in practice means that the SAA Boiler Code is used. The acts are generally administered by the State Departments of Labour and Industry, Departments of Mines, or Departments of the Interior, in which an inspection body is formed, in one case assisted by outside licensed boiler inspectors.

The new version of the SAA Boiler Code Part 1 of 1962 recognizes some foreign codes (British Standards, Lloyd's rules and the ASME Boiler Code) as acceptable.

BELGIUM

The "Règlement général pour la protection du travail" is issued as a royal decree, the definite contents, application and execution of which is to be decided by the administration. The "Règlement" contains a number of technical norms, especially regarding boilers and steam vessels, and also some norms for pressurized gas containers. There also exist, besides this decree, a number of technical codes published by the "Institut belge de normalisation" and by the "Association des industriels de Belgique".

Control of pressure vessels is exercised by the Inspection Division of the "Administration de la securité du travail et l'administration des mines". Periodic inspections are performed by private firms that have been licensed by the Government.

CANADA

In the Province of Ontario, where all Canadian nuclear reactors so far have been located, the "Boilers and Pressure Vessel Act" is the statute applying to boilers and pressure vessels. The Boiler Inspection Branch of the Ontario Department of Labour is normally responsible for the safety of boilers and pressure vessels. The Act requires that the Chief Inspector refer to publications of the Canadian Standards Association and of the American Society of Mechanical Engineers for rules as to approval of designs, manufacture, installation, inspection, testing and operation of boilers, pressure vessels and plants. The Canadian Standards Association's standard CSA B51-1960 is a "Code for the Construction and Inspection of Boilers and Pressure Vessels" which in its turn refers to the ASME and ASA Standard B31.1 codes as the standards governing the design, etc. of pressure vessels.

All provinces of Canada have made similar legislative provisions, although the degree to which the enforcement agencies are required to follow the CSA and other codes may vary from case to case.

DENMARK

The Danish Standards DS316, 317, 318, 320.1 and 320.2 published by "Dansk Standardis eringsräd" constitute the code for design, manufacture (welding) and inspection of boilers and pressure vessels. These codes will be revised in accordance with the Inter-Nordic INSTA recommendation concerning welding pressure vessels. The codes have an official status, and are under the control of the Danish Directorate of Labour Inspection.

FRANCE

The French norms for the design, manufacture and inspection of boilers and pressure vessels are laws and decrees collected in "Réglementation des appareils à vapeur" and "Réglementation des appareils à pression de gaz", published by the "Association des propriétaires d'appareils à vapeur et électriques" (GAPAVE). These laws and decrees constitute rules for safety and for official control of vessels, but they do not contain a code for calculation of vessel design. Such a code exists for unfired pressure vessels only, SNCT No. 1, issued by "Syndicat national de chaudronnerie et tôlerie". This code has no direct legal recognition.

The laws and decrees are administered by the "Ministère de l'industrie, Direction des mines".

GERMANY

For boilers a number of older codes will in the near future be replaced by codes and criteria drafted by the German "Dampfkessel- and Druckgefäss-Ausschuss" (DDA). These will be enacted by the Federal Government by way of statutory ordinances. The "Allgemeine polizeiliche Bestimmungen für Dampfkessel" specify tests and inspections for boilers.

The legal basis of the boiler codes is the Trading and Industrial Code. The codes themselves are considered as "Regeln der Technik" issued by virtue of the general administrative regulations for land-based steam boilers.

"Unfallverhütungsvorschrift Druckbehälter" contains safety rules for unfired pressure vessels and prescribes the tests and inspections to be performed. For design, manufacturing and installation it refers to accepted rules of engineering, in particular to the "AD-Merkblätter" established by the "Arbeitsgemeinschaft Druckbehälter".

The "Unfallverhütungsvorschrift Druckbehälter" is issued under a section of the "Reich Insurance Code", and also by practically all the trade associations concerned. The trade associations are obliged to make sure that the enforcement of it is supervised by technical inspectors.

Each province within the Federal Republic of Germany is responsible for the organization of the technical control. The competent authority in each "Land" is its Trade Supervisory Office, which normally works through or recognizes the "Technische Überwachungs-Vereine" (technical safety control services).

INDIA

The boiler code is the Indian Boiler Regulations, which are binding on all boiler owners and operators by virtue of the Indian Boilers Act. The Indian Boiler Regulations lay down the criteria for design, construction, commissioning, testing and inspection for boilers. Unfired pressure vessels come under the purview of the Indian Factories Act. There is at present no code for unfired pressure vessels, but the Factories Act stipulates inspection by the Inspector of Factories, who is empowered to take such action as is necessary to ensure safety. The Indian Standards Institution intends to establish an unfired pressure vessel code, but this would not have a mandatory status under the law.

There are Boiler Inspectorates and Factory Inspectorates in each state to enforce the Acts. These Inspectorates are guided by the Central Boilers Board and the Chief Advisor of Factories in the Ministry of Labour respectively.

ITALY

The National Association for the Control of Combustion (ANCC) has published circulars which constitute the code. Both the code and its legal standing are being revised. The ANCC has a central position concerning both writing and inspection.

JAPAN

The Regulation for Safety of Boilers and Pressure Vessels is a Ministry of Labour Ordinance which, together with the Ministry of Labour Notices, "Boiler Construction Code" and "Pressure Vessel Construction Code", applies to boilers and pressure vessels. Executive bodies are the Labour Standards Bureau of the Ministry of Labour and Labour Standards Offices and Labour Standards Inspection Offices in each prefecture. The inspections are performed by agents, the Boiler Association, the Safety Association or the Yasuda Fire and Marine Insurance Co. Ltd.

NETHERLANDS

"Grondslagen waarop de beeordeling van de constructie en het material van stoomtoestellen, damptoestellen en drukhouders berust" (Principles for the assessment of the construction and of the materials used in the construction of boilers and pressure vessels) together with a number of supplementary and explanatory publication sheets constitute the code, which is mandatory and has the standing of a regulation by virtue of the Steam Vessels Act and its decrees.

The inspectorate which enforces these laws is the "Dienst voor het Stoomwezen" (Boiler Inspection Department).

A decree is in preparation for unfired pressure vessels which, as regards design and construction, will conform closely with the regulations mentioned above.

NORWAY

The "Workers Protection Act" of 1956 establishes three codes as having status of law; "Standards and Rules for Power Boilers and Steam Pressure Pipe-Lines", "Material Specifications and Rules for Calculation of Strength", and "Standards for Welding of Boilers, Pressure Vessels and Steam Pressure Pipe-Lines".

The codes are enforced by the Boiler Inspection Department.

SWEDEN

A number of codes have been compiled and published by the Pressure Vessel Commission and the Welding Commission of the Royal Swedish Academy of Engineering Sciences, including a "Pressure Vessel Code", a "Boiler Welding Code" and a "Steam Boiler Code". There is a continuous effort to publish new codes for various special cases. These codes have no legal standing in themselves but the National Worker's Protection Board, which is the competent regulatory authority and which is represented on the above-mentioned commissions, has accepted and enforces them and they are in practice mandatory.

Inspections are carried out by competent, licensed private firms. There are no official rules regarding the neutral inspection of conventional pressure vessels during manufacture.

UNITED KINGDOM

The law (The Factories Act of 1961, section 32-39) gives a minimum of guidance on technical questions. Conditions of examinations are to be prescribed in regulations, which have not yet been written. Engineering codes exist, e.g. Lloyd's rules for boilers and unfired pressure vessels, rules of Associated Offices Technical Committee (AOTC) and British Standards for boilers and welded pressure vessels. For pressure-bearing components of reactor units general schedules of pre-commissioning inspection services and supplementary inspection and test requirements are contained in the Code of Associated Engineering Insurers. Design of reactor systems generally is based upon the requirements of British Standard 1500 for Fusion-Welded Pressure Vessels for use in the chemical or allied industries. Minimum supervisory procedure, detailed inspection procedures for materials, and methods of construction are specified for reactor systems in Lloyd's "Survey of pressure components for land-based nuclear installations – provisional requirements 1960".

The Factories Act imposes a statutory duty on the users of steam boilers and pressure vessels for the safety of their operation. Her Majesty's Factory Inspectors of the Ministry of Labour are responsible for the enforcement of the Act.

The codes have no legal standing in themselves but in reactor cases the Associated Engineering Insurers' Code and the AOTC Code have had regulatory status by virtue of a condition of Nuclear Site Licences. They have been enforced through the detailed design safety studies, supplemented by comprehensive and independent checks on the construction, inspection and testing of pressure vessels by Lloyd's or by Associated Engineering Insurers. The standards written by the British Standards Institution and the Royal Society for Prevention of Accidents also have no statutory force but they are prepared, to a large extent, in consultation with authoritative bodies and are generally accepted by industry.

The situation will probably change in the future as a result of the new Nuclear Installations Act of 1965 and the new B.S. 3915 standard of 1965.

USA

The code widely used for design, construction and initial inspection of boilers and pressure vessels is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code which since 1964 also includes a complete code section on nuclear vessels.

This code constitutes a legal document in those states and municipalities which chose to adopt it. Some authorities maintain their own separate codes which, however, generally are patterned after the ASME code.

Initial inspection is accomplished by states or insurance company inspectors who are certified by the National Board of Boiler and Pressure Vessel Inspectors, thereby providing acceptance by all states and authorities which have adopted the ASME code.

Requirements for recurring service inspections are in the hands of local and other enforcing agents, and administration or enforcement of such requirements vary from one state to another.

USSR

The following rules have been issued by the "State Committee of the Council of Ministers of the USSR on Safety Inspection in Industry and Mining":

(a) "Rules for Calculating the Strength of Steam Boiler Components 1956";
(b) "Rules for the Construction and Safe Utilization of Steam Pipes and Hot Water Pipes 1957";
(c) "Rules for the Construction and Safe Operation of Steam Boilers 1957".

These rules are binding on all ministries and departments.

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(6/-stg; F.Fr.4,-; DM 3,20)