Review

Evolution of Standardized Specifications on Materials, Manufacturing and In-Service Inspection of Nuclear Reactor Vessels

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Abstract: The cataloguing and revision of reactor pressure vessels (RPV) manufacturing and in-service inspection codes and their standardized material specifications—as a technical heritage—are essential for understanding the historical evolution of criteria and for enabling the comparison of the various national regulations, integrating the most relevant results from the scientific research. The analysis of the development of documents including standardized requirements and the comparison of regulations is crucial to be able to implement learned lessons and comprehend the progress of increasingly stringent safety criteria, contributing to sustainable nuclear power generation in the future. A novel methodology is presented in this work where a thorough review of the regulations and technical codes for the manufacture and in-service inspection of RPVs, considering the implementation of scientific advances, is performed. In addition, an analysis focused on the differences between irradiation embrittlement prediction models and acceptance criteria for detected defects (both during manufacturing and in-service inspection) described by the different technical codes as required by different national regulations such as American, German, French or Russian is performed. The most stringent materials requirements for RPV manufacturing are provided by the American and German codes. The French code is the most stringent with respect to the reference defect size using as a criterion in the in-service inspection.

Keywords: technical documentation; cataloguing; industrial heritage; standardized requirement; nuclear safety; historical advances

1. Introduction

On 11 December 1997, the Kyoto Protocol was approved, by which the United Nations Framework Convention on Climate Change was put into operation, containing the commitment of the industrialized countries to limit and reduce greenhouse gas emissions [1]. In addition, more recently, alarming reports from the Intergovernmental Panel on Climate Change have shown that climate change is a pressing matter that needs to be addressed, and in 2015, United Nations members agreed on keeping global warming below 2 °C through Nationally Determined Contributions [2]. Although the overall capacity of Renewable Energies has been gradually increasing and several countries have committed to ambitious climate targets since then, today's total energy supply is still primarily met by fossil fuels and the world is facing a prevailing, massive emission gap in reaching the Paris Climate agreement [3]. However, significant challenges for renewable energy development

include uncertainty in assessing local social impacts, stakeholder participation and social acceptance [4]. Global and regional trends indicate that energy demand may soon be supported by widespread deployment of renewable energy sources. However, the weather and climate-driven energy sources are characterized by significant spatial and temporal variability [5]. Renewable and nuclear energies are part of the energy sources that have contributed, and could participate even more in the future, to reducing carbon emissions [6]. For example, they can be exploited to generate "Green Hydrogen". Focusing on nuclear energy, the "classic" strategy is based on the use of nuclear as energy for electrolysis; but in the medium-long term, a more sustainable and smart approach could be founded on the use of thermochemical processes that require a direct coupling of a chemical plant to a nuclear reactor [7].

Present and future nuclear power plants (based on both fission and fusion reactions) require high levels of security, automation, and robustness to ensure protection from radiation [8]. Nuclear safety refers to all the technical dispositions and the organizational measures taken to prevent accidents or to limit their effects during the construction, operation, shutdown and dismantling of nuclear installations with ionizing radiation sources, as well as during the transport of radioactive substances [9]. In addition, the interactions between nuclear safety, security and safeguards must be taken into account [10].

Component failure is an issue of increasing concern as the current operating nuclear power plants reach the middle to latter portion of their design life and have accumulated service-related degradation [11]. The nuclear reactor pressure vessel (RPV) is one of the most critical components [12] to be considered when extending the operating period of a reactor. Due to harsh operating conditions, RPVs failed a serious accident could result during their operation [13]. The RPV in a light-water reactor (LWR) represents one of the key physical barriers preventing the release of radiation in case of an accident. Properly monitoring the behavior of the vessel materials is essential. In addition, mechanical test results often indicate that the reactor vessel materials can continue to operate safely beyond the reactor design life. Achievement of vessel integrity relies on many factors such as: a conservative design, balanced strength-ductility ratio, material toughness, high quality of welding and cladding operations, and the effectiveness and reliability of non-destructive testing [14]. The demands placed on RPV steels are severe, therefore, they must be manufactured in required dimensions and thicknesses, be of sufficient strength and toughness, show little deterioration under irradiation, and allow for the production of high-quality welds and be compatible with the cladding [15]. Thus, regulations that govern the operation of commercial nuclear power plants require conservative margins of RPV materials fracture toughness, both during normal operation and under accident scenarios [16]. Design codes provide a link between safety and design and a framework, in the form of design criteria and procedures necessary for performing structural integrity assessments, through which different operating conditions and occurrences can be assessed against recognized failure criteria [17]. Defects smaller than certain sizes can be accepted during the construction of nuclear power plants and later, in the in-service inspection of the nuclear power plant, defects beyond the defined acceptance range should be thoroughly monitored and analyzed [18].

While all nuclear equipment (immovable goods) is often well catalogued and analyzed, manufacturing and in-service inspection codes and their standardized material specifications (movable and intangible goods) have not been thoroughly studied based on a detailed evaluation of the technical heritage made up of the scientific and technical literature on the behavior of RPV materials [19], comparing regulated criteria. The radiation degradation of nuclear design materials limits the operational lifetime of all nuclear installations or at least decreases its safety margin. In addition, some investigations have not considered the growth of manufacturing defects due to radiation-induced processes in the context of material aging of RPV steels during operation [20]. In the first generation of nuclear power plants, restrictions for copper, nickel or phosphorous between other elements were not consider in the specifications. The surveillance programs and chemical composition

restrictions as well as the fracture mechanics methodologies were developed later and many of them are in progress nowadays. In addition, sensitivity of the non-destructive testing (NDT) increased with the development of the microelectronic and the safety and life- time calculations were driven through the advances in computing and the codes and standards could incorporate improved calculation methodologies. In the last pressurized water reactors (PWR) built (III generation), considerations such as a design life of sixty years, design conditions that allow simple maintenance, as well as short construction periods have been considered, compared to the constructions carried out decades ago. It should be noted that safety has been always considered paramount in the preliminary design stages.

In this work, a novel methodology is presented where a review of the state of the art is carried out, regarding the regulations of materials for the manufacture and in-service inspection of RPVs, tests to be performed on this type of materials both in their manufacture, and later, in the execution of the surveillance works, to ensure that they comply with the mechanical requirements established in the applicable regulations. In addition, an analysis focused on the differences between irradiation embrittlement prediction models and acceptance criteria for detected defects (both during manufacturing and in-service inspection) as described by the different technical codes according to different national regulations such as American, German, French or Russian is realized. Despite the relevance of the topic, there are no works in the scientific literature based on this approach; therefore, the analysis of evolution of standardized requirements and the comparison of regulations is crucial to implement learned lessons and understand the development of increasingly stringent safety criteria.

2. Methodology

The methodology of the performed analysis has been divided into 3 stages or phases (Figure 1). In the Phase A, an analysis of the technological literature on RPV design and manufacturing is performed. In the Phase B, an analysis of the technological literature on RPVs in-service inspection is carried out. Finally, in the Phase C a comparative analysis of manufacturing and in-service standardized requirements is presented.
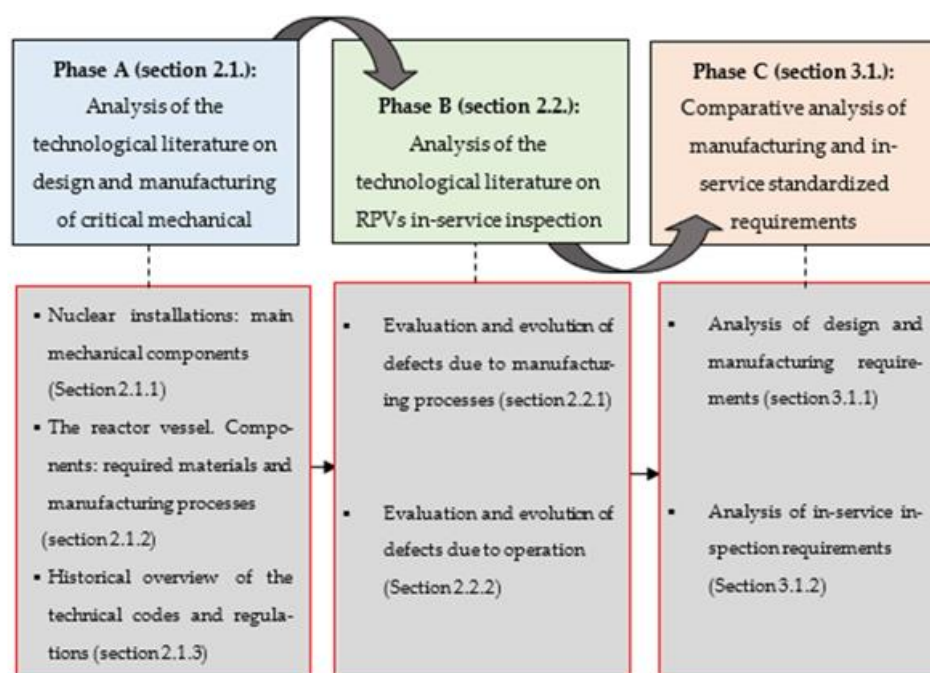
Figure 1. Methodology of analysis.

2.1. Phase A—Analysis of the Technological Literature on Design and Manufacturing of Critical Mechanical Components

During the design phase of critical mechanical components (including the RPV), a fundamental aspect to take into account is to determine which parts of these components are in contact with the primary coolant, which will determine the type of materials to be chosen in the design. Likewise, it is necessary to consider the working conditions of said components, highlighting among them those subject to internal pressure. In the case of the internals of the RPV, they are in direct contact with the primary coolant but are not subjected to a pressure gradient. This marks a differentiating element when it comes to addressing the mechanical design of these components.

2.1.1. Nuclear Installations: Main Mechanical Components

The main mechanical components of a nuclear power generation facility can be classified into active and passive structures, systems and components (SSC) [21]. Table 1 provides the active and passive components of a nuclear power plant classified by type.

Table 1. The active and passive components (SSCs) of a nuclear power plant classified by type [22,23].

Main Active Equipment and Components	Main Passive Equipment and Components
i Pumps (verticals and horizontals) ii Engines iii Valves iv Fans v Turbines (main turbine and pump guide turbines) vi Generators vii Emergency or Safeguard Diesel Generators viii Other SSCs (condenser, air compressors and regulators)	i Reactor Pressure Vessel ii Steel structures iii Reinforced concrete structures (buildings, containment, cooling towers and walls) iv Internal core components v Pipe vi Heat Exchangers vii Tanks viii Ventilation System ix Spent Fuel Pool x Fuel fixing elements xi Other SSCs (air ducts, accumulators, demineralizing filters, relief equipment and dehumidification towers)

According to the International Atomic Energy Agency (IAEA), most of the new reactors that have started to be built in recent years are PWR type plants [19]. Among all the mechanical components found in this type of nuclear power plant, the most unique components when compared to a conventional electrical energy production facility are mainly found in the primary coolant circuit, where the transport of the energy produced in the plants takes place.

The primary circuit and its components are in contact with aggressive fluids, such as the primary coolant that contains neutronic poisons such as boric acid, and they are exposed to high doses of irradiation. The reactor vessel that houses the nuclear fuel core is exposed to high doses of radiation and boric acid, so its design characteristics must take these aspects into account.

In general, in Light Water Reactors (LWRs) the most important mechanical components are:

- The RPV
- The pressurizer (in PWR plants)
- The steam generator (in PWR plants)
- The main coolant pumps
- The pipes that carry the coolant

The safe operational lifetime of RPVs depends on a number of factors, including design, chemical composition, microstructure, and mechanical characteristics of RPV steels and their in-service-induced change in properties, defect occurrence, and tolerance, as well as operating conditions [24].

2.1.2. The RPV. Components: Materials and Manufacturing Processes

RPV's main functions are to form the third containment barrier for fission products together with the reactor coolant system and to ensure the cooling of the fuel, keeping it submerged in the coolant. The vessel contains the core and its supports, the control rod bundles, and other accessories directly related to the core. The RPV is subjected to severe operating conditions such as pressure, temperature, radiation, stresses and an aggressive chemical environment. The specifications for design and construction of RPVs depend on the technology. The reactor vessel is typically a cylindrical low-alloy carbon steel shell with an internal austenitic stainless-steel cladding. The RPV consists of a cylindrical body and a hemispherical bottom and head.

The head is attached to the body by means of a closing flange that has a sealing gasket and tightening bolts. The flange is made up of a forged piece. The control rods are introduced through penetrations made in the hemispheres, being located in the upper hemisphere of the PWR reactors and in the lower part of Boiling Water Reactor (BWR). The nozzles for joining the vessel to the pipes of the reactor cooling system are located outside the core area, in order to reduce the level of neutron irradiation. Among the LWRs worldwide, the most prevalent are those of the PWR type. Henceforth, we will refer to PWR type reactor vessels. Figure 2 provides a simplified scheme of the primary loop of a PWR showing the role of technological evolution of materials for a continuous safety improvement.
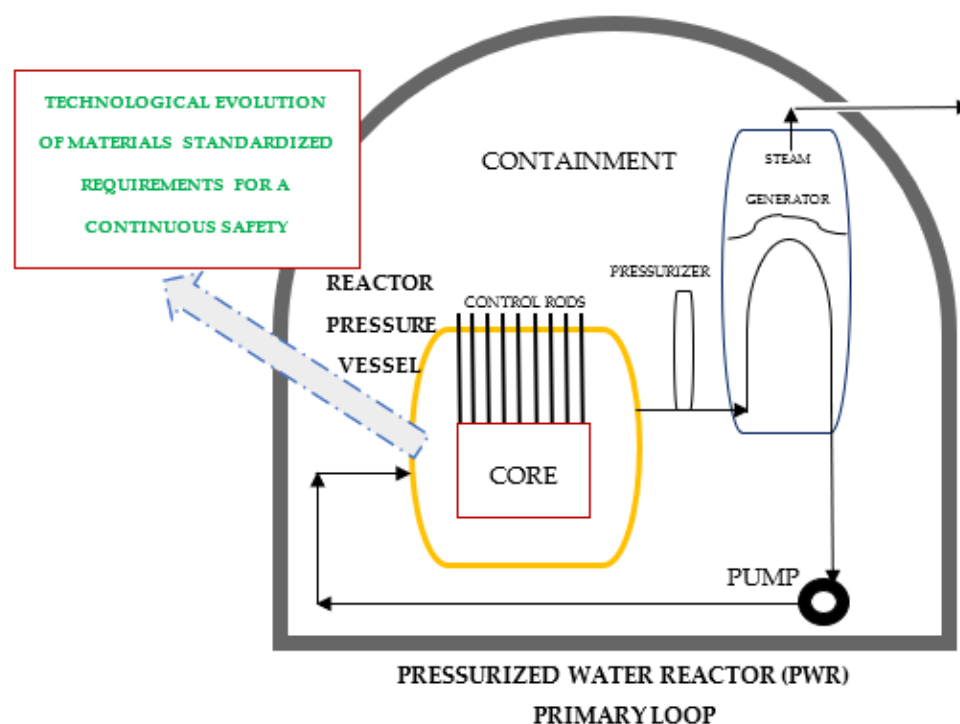
Figure 2. Simplified scheme of the primary loop of a PWR showing the role of technological evolution of materials for a continuous safety improvement.

Due to the size, weight and thickness of the elements that make up the vessel, it is necessary to use economically suitable and industrially available materials.

(a)   Required materials

The steels used in the construction of RPVs are low-alloy steels where the percentage of all the elements is below 5%. The low-alloy steels are cladded with an austenitic stainless steel. Strict control of impurities is necessary in these steels, since sulfur has a negative influence on resilience, while copper and phosphorus influence susceptibility to brittleness by neutron irradiation. In the case of RPVs, the shell material is cladded with

austenitic stainless steel to protect the vessel from corrosion. A suitable cladding process and material selection are crucial to avoid a pernicious effect like the hydrogenation. The materials to be used in the manufacture of nuclear reactors are addressed by internationally recognized codes, standards and specifications. Materials are not registered in the codes and standards, while there is not a wide experimental database of its properties, behavior, manufacture, processing and composition. Allowed materials must be in accordance with the corresponding design and manufacturing codes. The selected materials must withstand the effects of corrosion and must have sufficient strength to be used at the design temperature and pressure. An adequate selection of materials will ensure low initial costs (e.g., Capital Expenditures—CAPEX) as well as operational and maintenance costs (e.g., Operational Expenditures—OPEX). For example, the ASME B&PV code [25] does not recommend or suggest any type of material for any specific application. Instead, it indicates what materials are allowed and their requirements. Materials to be used in vessel construction are selected jointly by the equipment designer and the process manager. In all cases, some general features that the materials to be used in the manufacture of nuclear reactors must have are [23]:

- Adequate mechanical properties, such as tensile strength, ductility at working temperatures and good creep behavior.
- High thermal conductivity. To favor the elimination of heat from the core and to avoid the generation of internal stresses due to thermal fluxes with an anisotropic profile with respect to heat conduction.
- High resistance to thermal distortion. Due to the thickness of the walls of the pressure vessels, thermal stresses arise as a consequence of the temperature gradient along the thickness of the wall. A high value of the thermal stress is compensated, partially, by a high tensile strength.
- Low coefficient of linear expansion, or where appropriate, with a modulus similar to that of other materials. To avoid the generation of tensions between different components.
- Resistance to corrosion and compatibility with the other components of the reactor with which it must be in contact. The corrosion of the reactor materials, in addition to the associated consequences, such as loss of thickness, formation of cracks, etc., entails that the corrosion products are strongly radioactive and are carried by the coolant towards the outside of the reactor. The carbon steel used in the manufacture of the vessel is highly susceptible to corrosion due to the coolant in the primary circuit, so they cannot be in contact. This corrosion problem is solved by cladding the inside of the vessel with a thin layer of stainless steel. Protection by coatings in this manner is based on the so-called barrier technique, that is, placing an austenitic steel between the medium and the material to be protected that prevents the access of aggressive products from the environment to the material liable to be corroded. The choice of a specific coating depends on the nature of the material to be protected and the aggressiveness of the environment to which it is subjected. It is common to use grades according to the designation AISI 347 or AISI 308L because they do not precipitate chromium carbides at high temperatures. Another advantage of these materials is their weldability with Cr-Mo based materials. For the application in PWR type reactor vessels, the use of type 308L is recommended since it does not precipitate Chromium carbides at high temperature due to its low carbon content and because it presents an austenitic structure with a percentage around 6% ferrite, thus reducing hot cracking compared to AISI 347 steel, since the latter has an austenitic structure [26] which increases the risk of microcracks when conditions could include high temperatures (as when the material is welded or, in operation, in case of abnormal conditions), according to scientific studies have shown such as those published by Moorhead et al. [27] or Cui et al. [28]. Although carbon and low-alloy steels cladded with an austenitic alloy are used in the components of the primary reactor cooling circuit, in some cases, either as a result of repair or due to the existence of defects in the cladding, the base material is exposed to the cooling medium of the primary circuit.

For this reason, it is vitally important to monitor the RPV's mechanical properties, taking into account the possible contact of the carbon steel of the envelope with the reactor coolant, which may lead to boric acid corrosion [29].

- Ease of machining and weldability. A high machinability and adequate weldability are essential to correctly choose a material for the manufacture of an RPV [23]. Thus, usually the required C wt.% is lower than 0.15%.

- Adequate nuclear properties. In the RPV material located near the reactor core, an important aspect to consider in the selection of materials is how they influence the core physics and neutronics; for example, the material should have a small neutron capture cross section. It must be resistant to embrittlement as a result of irradiation and must have low induced radioactivity (from this point of view, the use of ferritic steels as base material is preferable to the more common austenitic steels).

To fulfill these requirements, low-carbon ferritic steels listed in design and manufacturing codes are used for the RPV shell. Although, equivalences between different standardized specifications of materials are usually established with respect to their main chemical and mechanical properties, the technological requirements described by equivalent specifications sometimes exhibit significant differences between them that could impact in their in-service behavior.

(b) Manufacturing processes

The RPV body and hemispheres are made up of rings that in turn are made up of curved and vertically welded sheets, although in some of the more recently manufactured vessels there has been an attempt to avoid welding in favor of complete forging pieces. So, henceforth, the work is focused on forging process as a process considered for the RPV manufacturing. Each forged piece must be manufactured according to the specification approved by the buyer, which typically includes the dimensions prior to tempering, the final dimensions, thicknesses that will withstand high mechanical stresses, and the locations of the specimens for conducting mechanical tests.

Specifically, ASME B&PV [25] requires that steel must be manufactured through a basic electric furnace process, except when secondary refining or remelting is necessary. Molten steel must be vacuum treated during the pouring phase of the casting, in order to remove gases trapped by the fluid, especially hydrogen [30]. After forging and before reheating, the forgings must be cooled to provide a complete transformation of the austenite. Preliminary heat treatment (PHT) can be used to improve machinability and complement other heat treatments. To improve mechanical properties, forgings must be heated to a temperature that produces an austenitic structure and then hardened in a suitable liquid medium by spray or immersion. In addition, tempering must be carried out after quenching at the subcritical temperature and maintaining this temperature for 1.5 h per inch of maximum thickness according to ASME B&PV code [25].

Finally, the forgings are welded. One of the most important operations during the manufacturing process of a pressure vessel is welding [31]. All steel grades are considered weldable under the proper conditions. The welding technique is of fundamental importance; welding procedures must be in accordance with approved methods for material grade in accordance with the requirements stablished in ASME IX: Welding and brazing qualifications [32] or in the required analogous code according to the applicable standards according to national regulations. After completion of manufacturing, nondestructive examination (radiographic, magnetic particle, liquid penetrant or ultrasonic examination) and hydrostatic test shall be performed in accordance with the requirements of ASME B&PV [25] Section III, Division 1 Subsection NB "Rules for construction of nuclear facility components-Class 1". After the pressure test of a vessel a volumetric examination shall be performed with equipment and techniques equivalent to those that are expected to be employed for subsequent in-service examinations.

2.1.3. Historical Overview of Technical Codes and Regulations

The United States Atomic Energy Act of 1954 opened nuclear technology to commercial applications. From the utility companies' perspective, the Atomic Energy Act of 1954 offered companies an opportunity to participate in nuclear development and gain experience with the technology [33]. Some concepts related to industrial standardization and, therefore, to the use in the manufacturing process of nuclear reactors are defined as follows [23]:

- Design and manufacturing code: Technical document, which clearly and concisely collects the rules and steps to be followed by the designer who is designing and manufacturing a specific structure or industrial good.
- Technical standard or specification: A technical standard is a specification of repetitive or continuous application whose observance is not mandatory except when a regulation or contract so determines. It is established with the participation of all interested stakeholders, which is approved by a recognized body, nationally or internationally. That is to say, this norm has a device character and not obligatory as it has a legal norm. Therefore, it is a technical document, issued by a standardization body, in which a set of conditions that a material, product or procedure must meet is specified.

Nuclear law must take its place within the normal legal hierarchy applicable in each nation. This hierarchy typically consists of four levels [34]:

- First: referred to constitutional level and International treaties or agreements.
- Second: statutory level, at which specific laws are enacted by a parliament in order to establish other necessary bodies and to adopt measures relating to the broad range of activities affecting national interests.
- Third: Regulations and highly technical rules to control or regulate activities specified by statutory instruments.
- Fourth: Non-mandatory guidance instruments.

Figure 3 provides the hierarchical pyramid [35] of nuclear safety regulation.

Figure 3. Hierarchical pyramid of nuclear safety law and regulation.

Thus, Table 2 shows the 4 levels of hierarchy applied to American, German, French and Russian regulations.

Table 2. Hierarchical levels according to US, German, French and Russian regulations [23,36].

| Hierarchical Level/Country	USA	Germany	France	Russia
Level 1	IAEA treaties and constitutional agreements			
Level 2	10 CFR 50	Atomic Energy Act	Act No. 2006-686 of	Federal law No. 170-FZ
Level 3	NRC guides	KTA and RSK guidelines	ASN guides	Safety guides
Level 4	ASME B&PV code ASTM standards ASME Code Cases Code interpretations	DIN, VDE, EN standards	RCC-M	PNAE

Technical standards or specifications are the documents that contain the manufacturing and in-service requirements related to the safety construction and operation of the critical components of a nuclear power plant. Henceforth, the focus of this work is centered on technical codes and the analysis of certain requirements related to the manufacturing and in-services stages of the RPV lifecycle. Standards and design codes are drawn up by national or international organizations of recognized prestige in the field of standardization. It is important to clarify that the regulatory standards must be located in a higher hierarchical range than the technical standards it regulates, so that there are no contradictions in the regulatory system [37]. A brief historical note [35] on the main design and manufacturing technical codes is developed as follows:

- ASME B&PV: The ASME B&PV code [25] is a set of standards, specifications, design formulas and criteria based on many years of experience, all of this applied to the design, manufacture, installation, inspection, and certification of vessels. under pressure. At the end of the 1700s, the use of boilers was becoming widespread and the need to provide guarantees regarding the safety of their designs was necessary since there were boilers that operated at pressures greater than atmospheric. In August 1907 in Massachusetts (USA) the Board of Boiler Rules was established, the first effective legislation on boilers in the USA, at the initiative of several insurance companies in order to reduce losses and claims. The committee that forms it is made up of engineers from all specialties and from all sectors in order to always keep it updated. Several attempts were made to standardize design criteria and calculations, but in 1911, due to the lack of uniformity for the manufacture of boilers, manufacturers and users of boilers and pressure vessels turned to the advice of the American Society of Mechanical Engineers (ASME) to correct this situation. Finally, in 1915, ASME published the first boiler code (the current Section I) in the United States. The codes were established to provide manufacturing methods, records, and also collect design data. Until 1930, the date of the first welded vessel, pressure vessels were riveted. The joints of the sheets were "overlapped" or strips of sheet metal were placed in the joints and they were perforated to be pierced with rivets. It was estimated that each rivet added pressure to the joint in a certain area of influence, thus guaranteeing the integrity of the equipment.
- KTA: The KTA safety standards [37] specify nuclear safety requirements to achieve the protection objectives established in the different radiation protection provisions, and in the "Safety criteria for nuclear power plants" in accordance with article 28, paragraph 3 of the "German regulations of radiological protection and guidelines in the event of an accident (edition of 18 October 1983)". Currently, the KTA standards program consists of 98 different standards. The safety criteria require the establishment of a comprehensive quality assurance system for the manufacturing, construction and operation of nuclear power plants. The KTA standards require the application of a large number of conventional standards (in particular, the DIN standards).
- RCC-M: In 1978, CEA, EDF and NOVATOME decided to draw up a code with design and construction rules for components of light-water nuclear power plants. The RCC-

M code was published for the first time in June 1985 and again edited in May 1993. The last edition of the code was published in 2007 [38]. The scope of RCC—M exclusively covers mechanical components of nuclear power plants, considered relevant with respect to the safety and availability of the plant. These components are: tanks, supports, containers, vessels, reactor internals, heat exchangers, pumps, valves, pipes and mechanisms for handling and controlling the reactor.

- PNAE: The first specific Russian code for the design of nuclear pressure vessels was published by "Metallurgy Press" in 1973 and was approved by the state committee for nuclear energy and its regulatory body, under the name "Gosgortechnadzor". Subsequent editions of the code have added the experience gained and reflected in the editions of the ASME B&PV code.

The design and manufacturing requirements of RPVs are specified in ASME III Div.1 (Figure 4).
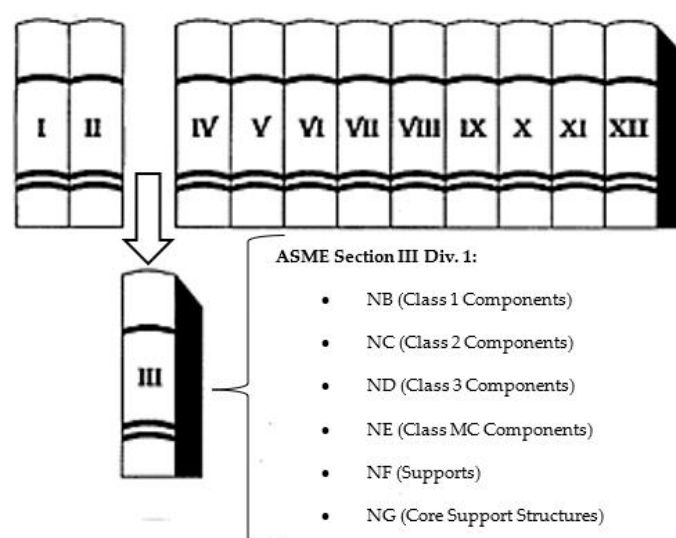
Figure 4. Structure of section dedicated to RPV manufacturing according to ASME B&PV code.

The extensive experience deposited in the development of the ASME B&PV code [25] since its creation at the beginning of the 20th century has led it to be a reference standard in the design of most of the world's nuclear power plants. Nowadays the majority of reactors in operation have been designed based on the American code. For this reason, many standards of recognized prestige like the French among others have been based on the ASME design methodology and manufacturing requirements. The equivalent sections in German, French and Russian codes are indicated in Table 3.

**Table 3.** Structure of codes related to RPV manufacturing [17,39,40].

ASME Section III Div. 1	KTA	RCC-M	Russian Code (GOST)
NB (Class 1 Components)	KTA 3201.2 and KTA 3211.2	Section I - B	
NC (Class 2 Components)	KTA 3211.2	Section I - C	
ND (Class 3 Components)	KTA 3211.2	Section I - D	Groups A, B and C of equipment and pipelines
NE (Class MC Components)	KTA 3401.2	Section I - P	
NF (Supports)	KTA 3201.1, KTA 3201.2 and KTA 3205.1	Section I - H	
NG (Core Support Structures)	KTA 3204	Section I - H	

Thus, by way of example, in the case of the American nuclear legislation, the standards or sections of design codes (ASTM and ASME B&PV) are only mandatory when they are referred to as mandatory in the set of CFR laws (Code of Federal Regulations) and in the

American regulatory standards (Regulatory Guides or Regulatory Rules) issued by the U.S. Nuclear Regulatory Commission. In the case of German legislation, the regulation is developed by ordinances and by the radiation protection law itself, being the KTA standards, the RSK guides and the DIN, EN, or EN ISO standards regulated by the above [35].

2.2. Phase B—Analysis of the Technological Literature on RPVs In-Service Inspection

It is recommended, mainly in projects of a certain size (from the design and manufacturing up to the monitoring of components in-service operation), that at the beginning of each project the order of prevalence of the technical information to be handled is established. This action is the responsibility of the customer. If this action is not carried out, it is possible to overlook a document which should have been considered. Figure 5 provides the relationship between technical codes, applicable specifications and their standardized requirements.
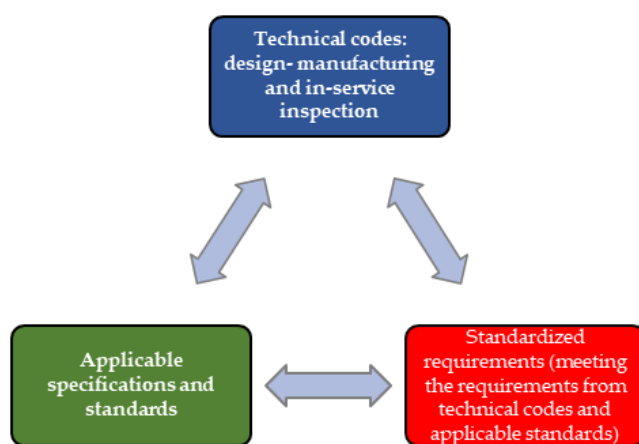
Figure 5. Relationship among technical codes, specifications and their standardized requirements.

Failure in materials, components, and assembly has demonstrated that the current techniques of fabrication and in-service inspection, sometimes, are not sufficient alone to assure consistent reliability in critical components. Flaws and inhomogeneities occur even when using the best processes and properly controlled procedures. Thus, an adequate and well-integrated nondestructive testing program is necessary to assure the quality level required in a nuclear reactor pressure vessel. The historical nondestructive methods used in the fabrication of reactor pressure vessels are: visual, X-ray and gamma radiography, ultrasonic, magnetic particle, and liquid penetrant testing [41]. RPV in-service inspection requirements are indicated in Section XI of the ASME code and the equivalent sections in KTA, RCC-M or PNAE codes, among others.

The importance of proper flaws detection and sizing using nondestructive examinations (NDE) has always been considered, for this reason many research programs have been developed to improve the inspection techniques. In addition, the U.S. nuclear regulatory commission (NRC) required through the law 10 CFR 50, an independent qualification of NDE procedures, equipment and personal in accordance with ASME B&PV [25] Section XI, Appendix VIII, in order to assure that NDE system is capable of achieving a specific level in the detection reliability and sizing accuracy. In 1995 the first issue of the document "European Methodology for Qualification of Non-Destructive Testing" was published by The European Network for Inspection and Qualification (ENIQ), which harmonized the position of the Europeans countries with regard to the subject of qualifying validating inspection systems. Since then, several new revisions have been issued, being the last one issued in 2007 [42].

2.2.1. Evaluation and Evolution of Defects due to Manufacturing Processes

The imperfections found in structures may be classified into global or local types: the first ones are related to deviations of the "perfect structure" that extend to a large part of the component (e.g., out-of-roundness of the cross-sections or thickness variations), whereas the latter ones are related to those deviations that are restricted to a small part of the component (e.g., small dimples or small indentations on the shell surface, welding defects, small corroded areas, etc.) [43]. Heat treatment technology (normalizing and tempering) for large forgings is based on the experimental investigations and experiences of actual operation in addition to heat treatment theory. The most relevant forging defects to be considered are tearing around inclusions, thermal cracking, cracks to overloads and surface imperfections. Welding cracks are typically one of the following general types [44]:

- Cold cracking, also called heat-affected zone (HAZ) cracking, occurs during cooling when the stress of solidification causes the weaker solid metal adjacent to a weld bead to crack.
- Hot cracking takes place as a result of the strains set up during welding and occurs in thin films of nonmetallic segregates or by segregation of alloy elements, both of which lead to intergranular surfaces that solidify after the rest of the weld metal.
- Hydrogen induced cracking: flake, fisheye and shatter crack are well known hydrogen-induced defects. These defects are generated in the production and manufacture of the component and not through ageing [45]. Recently, problems in large forgings due to hydrogen have decreased tremendously thanks to improvements in vacuum treatment technology [15]. After preliminary welding heat treatment (PWHT) a martensitic layer can be created along the austenitic/ferritic interface and high residual stresses can be cumulated. If, for any reason, hydrogen is introduced into that area, it may cause cold cracking and separation of the cladding and base metal. Hydrogen can be introduced in subsequent welding operations on the cladding that come to thermally affect the ferritic steel.

There are no published reports of incidents in nuclear vessel fabrication of the first two forms, indicating that the consumables and procedures normally selected in European and American nuclear fabrication shops have adequate resistance to these types of cracking [15]. Because of the significant difference in the thermal expansion coefficients of the cladding and the base metal, cladding induced stresses can be generated. Even after PWHT, residual stresses of yield magnitude remain in the cladding and the HAZ at ambient temperature. The cladding induced stresses have a significant influence on small defects near the inside surface of a pressure vessel [46]. Figure 6 shows the different layers that make up the RPV (shell, 1st and 2nd layer of cladding and the typical location of these sub-cladding defects (cracks).
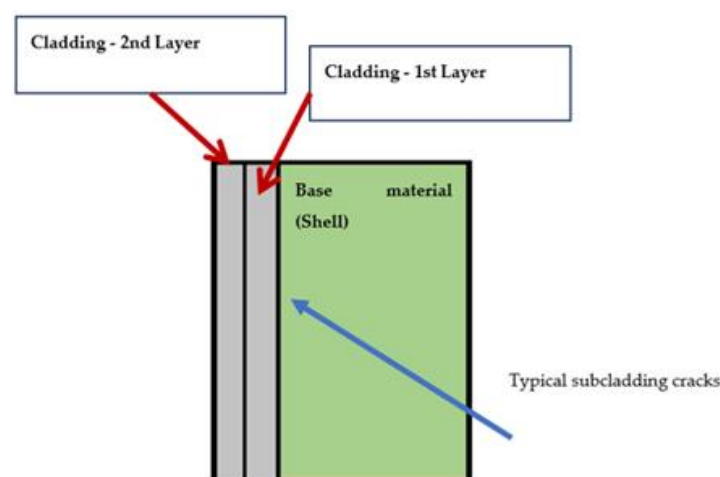
Figure 6. Distribution of cladding layers and location of typical sub-cladding cracks in the shell material.

Several cases have been reported where cracks in RPV have been detected. In some nuclear power plants (NPP) have been found cracks or other defects in several parts like top head nozzles or bottom-mounted nozzles (BMN) and shell. Leaks in reactor vessel top head were discovered in some U.S. PWR and French NPPs due Primary Water Stress Corrosion Cracks (PWSCC) from the outside of the nozzle above the J-groove weld which produced boric acid deposits on the vessel head near the nozzles, as a result a corrosion of the low-alloy steel head material. The more representative experience has been Davis Besse NPP in 2002. As consequence, an augmented inspection plans have been implemented in PWR fleet in accordance with new regulations standards (i.e., code cases), and others action like the replacement of head vessel by others with material more resistance to PWSCC (i.e., alloy 690) have been performed. Other operating experience like cracks detected in BMN (in 2003 at South Texas Project Unit 1) or hydrogen flakes in a vessel shell of Doel 3 and Tihange 2 NPPs-Belgium) has demanded new analysis and implementing the additional inspection plans. For example, the news about the high number of relevant quasi-laminar indications detected in the forged core shells at Doel 3 and Tihange 2 (hydrogen flakes), raised questions in Germany regarding inspection during manufacturing as well as during in-service inspection [47]. Following forging, and other high-temperature operations, hydrogen can collect at metallurgical inclusions. The growth of defects due to radiation-induced processes, especially radiation-enhanced diffusion and radiation induced segregation is also necessary to be considered in the context of material ageing of RPV steels during operation [19].

As important issue due to industry operating experience and research developments has been to develop within the framework of ageing management for a log-time operation (LTO) a guideline to develop engineering and inspection programs to manage ageing in PWR and BWR internal vessels. Table 4 summarizes the most typical defects found in RPVs generated during manufacturing and in service.

Table 4. Most typical defects found in RPVs generated during manufacturing and in service.

Main Manufacturing Defects in the RPV Shell	Main In-Service Defects in the RPV Shell
Shell manufacturing defects: • Tearing around inclusions • Thermal cracking • Cracks due to overloads • Surface imperfections	Irradiation Embrittlement that depends mainly on: • Chemical composition (mainly Cu, P and Ni wt.% contents) • Neutron Flux ($n/cm^2 \cdot s$) • Neutron Fluence (n/cm^2) • Irradiation temperature
Welding defects: • Cold cracking and hot cracking	Corrosion of materials and corrosion erosion, stress corrosion and corrosion – fatigue combined processes.
Hydrogen defects: • Hydrogen flaking • Fish eyes and shatter cracks	Evolution of manufacturing defects (typically cracks) affected by in-service conditions

2.2.2. Evaluation and Evolution of Defects due to Operation

The integrity of the RPV depends on material properties and their time dependent degradation (ageing). The major degradation mechanisms are neutron embrittlement, low cycle fatigue and possible corrosion attack [48]. Historically, the structural factors contributing to vessel failure potential are [49]:

• Design deficiencies
• Fabrication flaws
• Service deterioration of vessel materials:
    – Loss of structural integrity
    – Changes in mechanical properties

The most important process of neutron interaction is that produced by fast neutrons (>0.5 MeV), which modify the mechanical properties of materials by colliding with and distorting atoms in the crystal lattice. This distortion results in hardening, an increasing of the yield point and elevation of the ductile-brittle transition temperature (ΔT_{DBT}). On this basis, structural materials are usually divided into three categories. The first includes those materials whose effective neutron cross section is low enough so that they can be used in natural uranium reactors; the second includes materials with an intermediate cross section, and which are suitable for use in enriched uranium reactors; in the third category are materials whose effective cross section is so high that their use is restricted to fast reactors. In the present case, second category materials are considered.

Table 5 shows a simplified Failure Mode and Effects Analysis (FMEA) that can help understand main probable failure modes and the effects of their occurrences [50].

Table 5. Main mechanisms of in-service RPV degradation and its impact on reactor safety [23,51].

Degradation Mechanism	Failure Mode and Effect Analysis	
	Effect on the Structural Component (Failure)	Impact to the Safety Function (Effect)
Corrosion Corrosion - erosion Stress corrosion cracking	Thickness thinning Augmented risk of brittle fracture	Possible loss of full structural capacity against mechanical stresses generated by severe operating conditions
Fatigue and corrosion-fatigue	Fatigue crack nucleation	They would produce a loss of mechanical integrity in the material, which could generate a catastrophic failure
Irradiation embrittlement	Increase in the ductile-brittle transition temperature of the material with structural function	Loss of toughness, favoring breakage and increasing the probability of catastrophic failure

For the typical operating conditions of a nuclear reactor, chemical composition more strongly influences [52] the process of neutron irradiation embrittlement than neutron fluence [53] and irradiation temperature [54].

During long-term operation, the fast neutron fluence (E > 0.5 MeV) causes the ferritic steel of the RPV to become susceptible to brittle fracture, especially shell and weld materials in the beltline region corresponding to the reactor core. The embrittled vessel shell may fracture due to a preexisting fabrication flaw which could lead to a through-wall crack [55]. To experimentally determine the value of ΔT_{DBT} and, therefore, the fracture toughness of irradiated material, surveillance capsules are included in the vessel between the core and the wall.

These capsules contain samples of the material of the vessel, both welds and heat affected zone as base material. These surveillance capsules are removed periodically in order to test the specimens, to predict in advance if the material is affected by irradiation embrittlement [56]. A prediction of radiation shift in design stage brittle fracture analysis shall be based on empirical correlation between steel alloying elements, impurities and fast neutron fluence [57]. In order to understand, in advance, the evolution of this fragility, so-called Surveillance Programs are carried out, the objective of which is to quantify the degree of fragility achieved in advance.

This surveillance program conforms to Appendix H of the law 10CFR50 [58] in the case of American nuclear power plant license bases, and to the standard KTA 3203 [59], under German license conditions [60]. In essence, the program consists of placing capsules near the wall of the vessel and at the height of the center of the core (Belt—Line), during the construction phase, capsules containing in their interior specimens of the same material used in the manufacture of the vessel (i.e., samples of base metal, weld and heat affected zones in welds) and with the same thermo-mechanical treatment of the vessel, as well as pure nickel, pure copper, niobium and iron wires, as well as other elements that can

be used to evaluate the neutron fluence. wires, which serve as dosimetry standards to determine with accuracy of the neutron fluence in that area [23].

Table 6 provides the most recognized and used analytical models and constraints for degradation prediction.

Table 6. Most recognized and used analytical models or constraints for degradation prediction in RPV material [19,56,61].

Prediction Model	Description and Formulation
R.G. 1.99 Rev.2 [62]	R.G. 1.99 Rev.2 proposes a model for calculating the ductile-brittle transition temperature shift depending on the copper and nickel content and neutron fluence, according to Equation (1): $$\Delta T_{DBT} = (CF) \cdot f^{(0.28 - 0.10\ log\ f)} \qquad (1)$$ In Equation (1), *CF* is the chemical factor provided by R.G. 1.99 Rev. 2, which is a function of copper and nickel content in wt%, and *f* is the neutron fluence in n/cm^2.
NUREG CR-6551 [63]	$$\Delta T_{DBT} = SMD + CRP \qquad (2)$$ $$SMD = A\ exp\ [C_{Tc}/(T_c + 460)]\ [1 + C_P\ P]\ (\varphi t)^{\alpha} \qquad (3)$$ $$CRP = B\ [1 + C_{Ni}\ Ni^{\eta}]\ F(Cu)\ G(\varphi t) \qquad (4)$$ To obtain the CRP contribution, it is necessary to calculate the *F(Cu)* (Equation (5)) and the *G(φt)* (Equation (6)) parameters. $$F(Cu) = \{0,\ Cu \leq Cu_{th}; (Cu - Cu_{th}),\ Cu > Cu_{th}\} \qquad (5)$$ $$G(\varphi t) = +\ tanh\ \left\{ \left[\log\left(\varphi t + C_t\ t_f\right) - \mu \right]/\sigma \right\} \qquad (6)$$ Phosphorus content and the considered manufacturing process -forging or rolled- was included in the model with respect to R.G. 1.99 Rev.2 [64]
ASTM E-900 [65]	$$\Delta T_{DBT} = SMD + CRP + Bias \qquad (7)$$ Where, $$Bias = \{0, t_i < 97000\ h; 9.4,\ t_i \geq 97000\ h\} \qquad (8)$$ and t_i is the irradiation time
RCC-M [38]	$$\Delta T_{DBT} = 22 + 556(\%Cu - 0.08) + 2778(\%P - 0.008) \cdot \left(\frac{f}{10^{19}}\right)^{1/2} \qquad (9)$$
KTA 3203 [59]	Some constraints are imposed to reduce susceptibility to irradiation embrittlement (using as a reference the R.G. 1.99 Rev.2 model): $$Cu \leq 0.15\%,\ \forall\ 0 < Ni \leq 1.1$$ $$Ni \leq 1.1\%,\ \forall\ 0 < Cu \leq 0.15 \qquad (10)$$
GOST/PNAE	$$\Delta T_{DBT}\ (^{\circ}C) = (575 \cdot (P + 0.1\ Cu) + 20) \cdot (18\ .\phi)^{1/3} \qquad (11)$$

Note: ∀—Logical constant interpreted as "given any" or "for all".
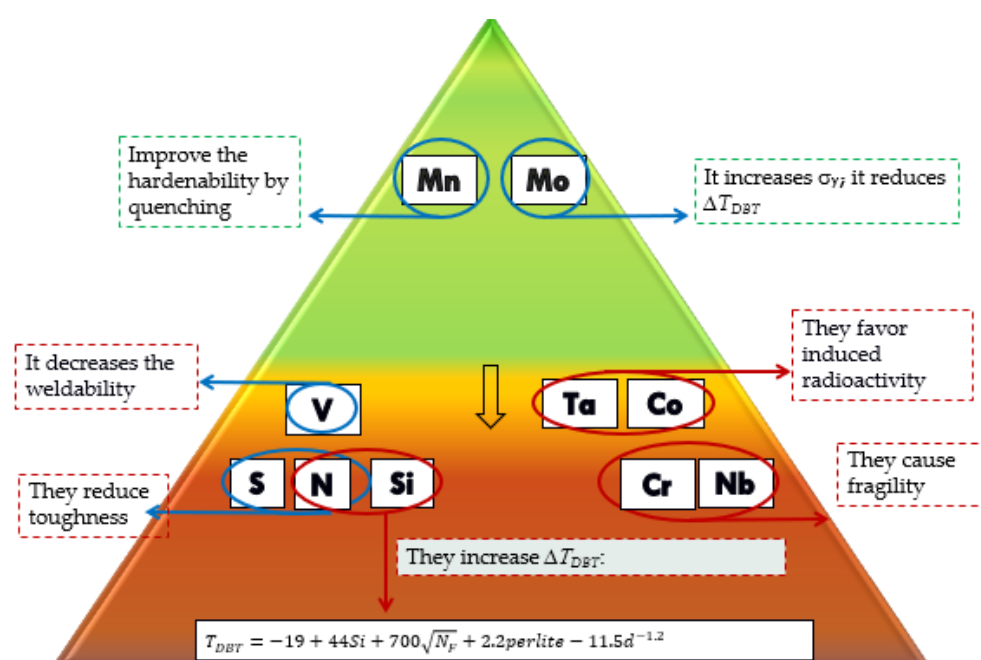
Using the models presented in Table 6, copper and nickel thresholds for the American models (R.G. 1.99 Rev.2, NUREG/CR-6551 and ASTM E 900-02), French (RCC-M), German (KTA 3203) and Russian (PNAE) are shown in Table 7.

On the other hand, Figure 7 provides the impact on in-service behavior of other elements (not copper, nickel and phosphorous) ranked by their influence.

Table 7. Copper and nickel thresholds obtained on the basis of R.G. 1.99 Rev.2, NUREG CR-6551 and ASTM E900-02 and RCC-M using the maximum ΔT_{DBT} specified by KTA 3203 to avoid additional safety calculations, i.e., 40 °C for a neutron fluence of 1×10^{19} n/cm^2 [56,61].

R.G. 1.99 Rev.2	NUREG/CR-6551	ASTM E 900-02	RCC-M	KTA 3203	PNAE
Cu $\leq$ 0.25, $\forall$ 0 < Ni $\leq$ 0.2 and $\forall$P (wt%) Cu $\leq$ 0.20, $\forall$ 0.2 < Ni $\leq$ 0.4 $\forall$P (wt%) Cu $\leq$ 0.16, $\forall$ 0.4 < Ni $\leq$ 0.6 $\forall$P (wt%) Cu $\leq$ 0.14, $\forall$ 0.6 < Ni $\leq$ 0.8 $\forall$P (wt%) Cu $\leq$ 0.13, $\forall$ 0.8 < Ni $\leq$ 1.2 $\forall$P (wt%)	0.15 $\forall$ Ni wt% < 0.6	0.15 $\forall$ Ni wt% 0.2 only if Ni wt% < 0.4	0.05 $\forall$ Ni wt% and P < 0.02 wt.%	Cu $\leq$ 0.15%, $\forall$ 0 < Ni $\leq$ 1.1 Ni $\leq$ 1.1%, $\forall$ 0 < Cu $\leq$ 0.15	0.06 $\forall$ Ni wt% and P < 0.02 wt.%

Note: $\forall$—Logical constant interpreted as "given any" or "for all".

Figure 7. Impact on in-service behavior of other elements (not copper, nickel and phosphorous) ranked by their influence.

Manganese plays a key role as it enhances tensile strength and maintains ductility [66]. Likewise, a study has shown [67] that the presence of Mn reduces the influence of the material's manufacturing method (forging or rolling) on its behavior against neutron irradiation. Molybdenum is a complex carbide former, such as $Fe_{21}Mo_2C_6$; this effect improves the elastic limit, the maximum tensile strength and reduces the ductile-brittle transition temperature [68]. Manganese can combine with sulfur to form soft manganese sulfide (MnS). This prevents the formation of iron sulfide at the grain edges, which would produce brittleness [69]. ASME B&PV code [25] does not specify maximum vanadium content for rolled material SA-533, when it would be necessary, since vanadium can reduce the weldability. Likewise, ASME B&PV code [25] does not specify contents in tantalum and cobalt, these elements being of great importance, since they help the material to become a source of production of γ radiation, once the reactor has stopped, due to the induced radioactivity that is generated, produced by neutron bombardment. This can be of great importance, for example at reactor shutdowns for refueling [70]. Cobalt also has a hardening action [71], so its content must be controlled to ensure adequate ductility of the material, also taking into account embrittlement by neutron irradiation. Lee and Kim [72] experimentally verified how the presence of dissolved MnS at a crack end increases the crack growth rate by conducting fatigue tests, in high temperature water. This argument reinforces the importance of reducing the sulfur content to minimum levels. On the other hand, nitrogen reduces ductility and toughness [73] and provides brittleness to steels. It is

usually found in a combined form, forming nitrides. In addition, it is necessary to keep the silicon content to the minimum possible [74] to obtain an adequate toughness of the material [75], since it influences the ductile-brittle transition temperature (ΔT_{DBT}) [76]. Chromium content is limited, because it can favor the formation of chromium oxides that precipitate at the grain boundary, causing brittleness, as observed by Rosario and Villacorta [77,78]. The presence of niobium increases the hardness of carbon steels [78], so it seems appropriate that the ASME B&PV code [25] restricts its content to a value of 0.01%, since a ductile behavior is required at high temperatures. Both ASME B&PV code and KTA [37] in their SA-533 Gr. B and DIN 20MnMoNi55 specifications, respectively, do not specify Niobium content; the presence of this element being unwanted since it increases the hardness of the material.

3. Results

### 3.1. Phase C—Comparative Analysis of Manufacturing and In-Service Standardized Requirements

Given the experimental restrictions provided by [79–83]: Cu wt.% max = 0.10, P wt.% max = 0.02 and Ni wt.% max = 1.00, a comparison of requirements on the most common materials used in RPV shell manufacturing is performed (Section 3.1.1).

3.1.1. Analysis of Design and Manufacturing Requirements

Once analyzed, the influence of the most relevant standardized requirements, Cu, P and Ni wt.% are evaluated quantitatively using, firstly experimental limits [79–83]. In addition, it is also applied the calculation schemes provided by R.G. 1.99 Rev.1 using the restrictions of maximum ΔT_{DBT} established by KTA 3203 and the permissible variation on chemical analysis as described by ASTM A-788 (Figure 8) and RG DG-1070 (Figure 9). Thus, this can be valid for using the materials for the shell or for mechanical accessories (that could be purchased also as a commercial-grade).
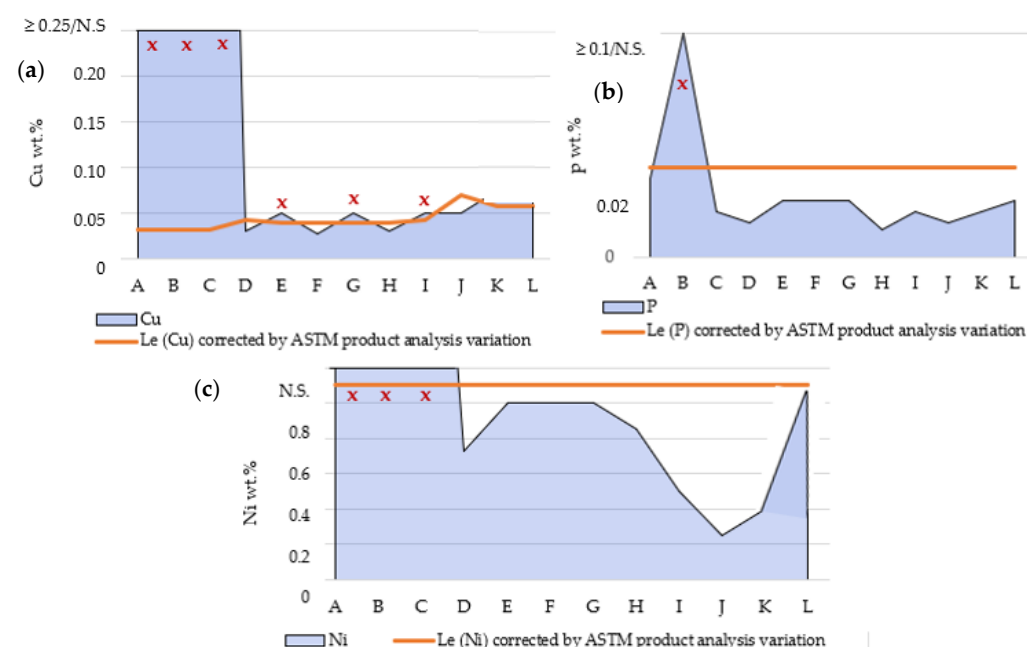
Figure 8. Cu (**a**), P (**b**) and Ni (**c**) wt.% compared with the experimental limits corrected considering maximum permissible variation on product chemical analysis according to ASTM A-788. Note: A—ASTM A 212B; B—ASTM A 302B; C—ASTM A 543 B; D—ASME SA 533 Grade B Cl.1; E—ASME SA 508 Grade 2; F—DIN 22NiMoCr37; G—ASME SA 508 Grade 3; H—DIN 20MnMoNi55; I—RCC 16 MND5; J—ASTM A 336 Grade F22V; K—WWER 15Kh2MFA; L—WWER 15Kh2NMFA. N.S—Not specified. X: out of experimental limit (Le (Cu), Le (P), Le (Ni)) or because it is not specified. Le (Cu), Le (P), Le (Ni) are the experimental limits [79–83] indicated in some historical research works that often has been considered in some specifications of NPPs.
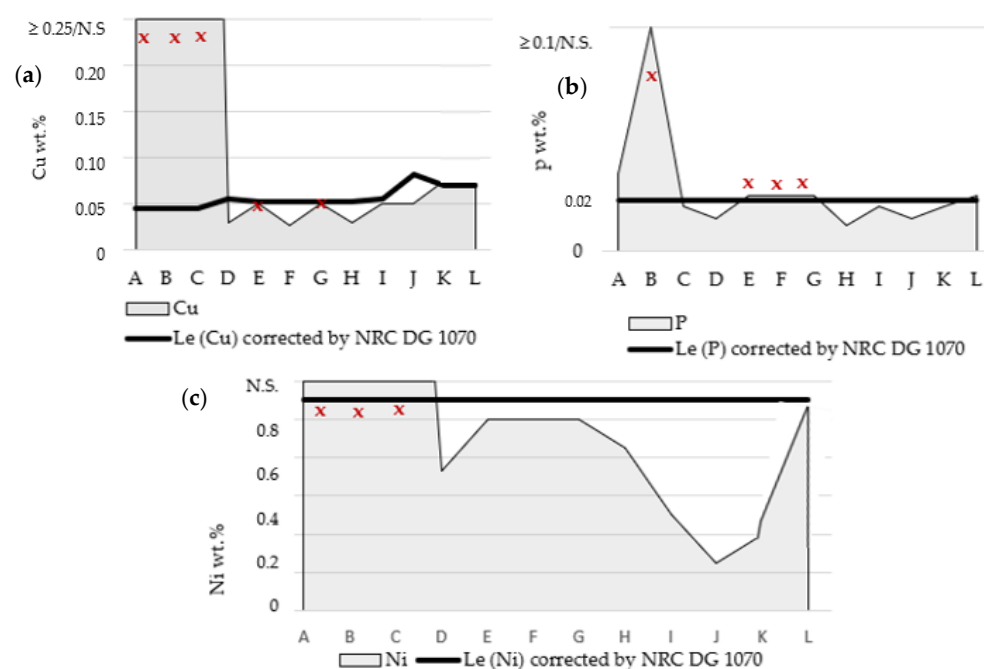
Figure 9. Cu (**a**), P (**b**) and Ni (**c**) wt.% compared with the experimental limits corrected considering maximum permissible variation on product chemical analysis according to R.G. DG-1070. Note: A—ASTM A 212B; B—ASTM A 302B; C—ASTM A 543 B; D—ASME SA 533 Grade B Cl.1; E—ASME SA 508 Grade 2; F—DIN 22NiMoCr37; G—ASME SA 508 Grade 3; H—DIN 20MnMoNi55; I—RCC 16 MND5; J—ASTM A 336 Grade F22V; K—WWER 15Kh2MFA; L—WWER 15Kh2NMFA. N.S—Not specified. X: out of experimental limit (Le (Cu), Le (P), Le (Ni)) or because it is not specified. Le (Cu), Le (P), Le (Ni) are the experimental limits [79–84] indicated in some historical research works that often has been considered in some specifications of NPPs.

U.S. NRC regulatory guide DG-1070 [84] provides tolerances for chemical elements analysis applied to simple metallic commercial grade items that can be used in the reactor environment.

Figures 8 and 9 shows how some materials requirements do not meet some more stringent requirements used in the customary materials selection tasks for NPP. Basically, the restrictions take place when the materials specification does not include the wt.% content in Cu, P or Ni as is the case with the specifications ASTM A 212B; B—ASTM A 302B; C—ASTM A 543 B. In Figure 8 has been considered the maximum permissible variation on product chemical analysis according to ASTM A-788 for RPV base materials and in Figure 9 the maximum permissible variation on product chemical analysis according to R.G. DG-1070 for mechanical (metallic) accessories (that could be purchased also as a commercial-grade).

### 3.1.2. Analysis of In-Service Inspection Requirements

On the other hand, Figure 10 relates ΔT_{DBT}, neutron fluence and chemical composition of representative material candidates (selected from the results shown in Figures 8 and 9) to fulfill the criterion of maximum ΔT_{DBT} (from 1 to 5×10^{19} n/cm^{-2}) to avoid additional safety calculations according to KTA 3203 as is established in the usual materials properties' surveillance performed within the in-service inspection programs.

In the Figure 10, the numbers in red indicated the ΔT_{DBT} as a function of the Neutron fluence (from 1 to 5×10^{19} n/cm^{-2}) and of the materials considered in this example, showing the fulfillment with the criteria provided by KTA 3203 to avoid safety calculations.

WWER 15kh2MFA, DIN 20 MnMoNi55, DIN 22NiMoCr37, A 533 Gr. B meet the KTA 3203 [59] maximum ductile-to-brittle transition temperature shift for a neutron fluence from 1 to 5×10^{19} n/cm^{2}. In addition, these materials also meet the experimental thresholds

corrected by the maximum allowable variation according to ASTM A-788 [29]. Nevertheless, considering additional restrictions imposed by these standardized specifications such as those established from the application of the prediction models (as it shows in Table 6) is possible that the remaining materials could operate under the reactors' harsh conditions. Figure 11 exhibits the fulfillment of R.G. 1.99 Rev.2 and KTA 3203, also showing the fulfillment when is considered the maximum permissible variation on product chemical analysis according to ASTM A-788 and R.G. DG-1070 as provided in Figures 8 and 9.
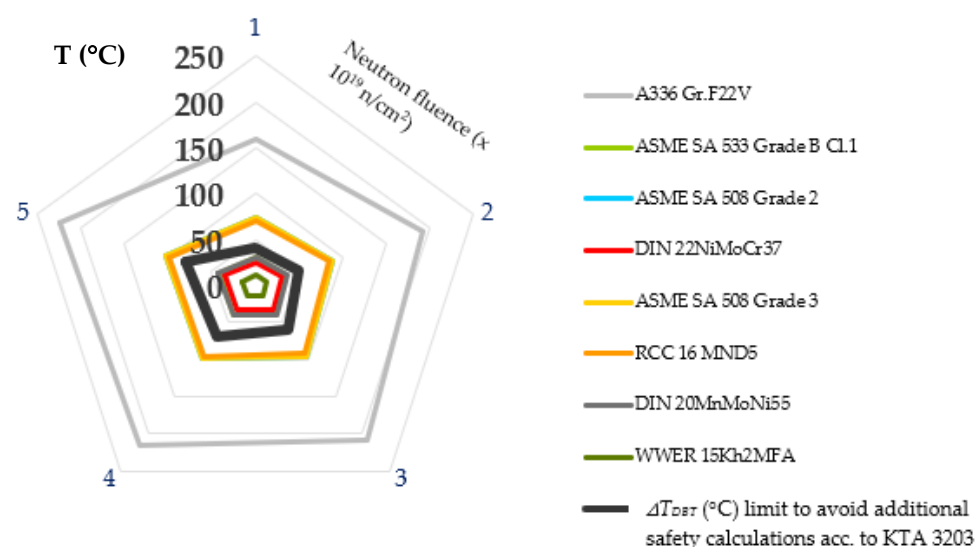
Figure 10. Relation between ΔT_{DBT}, neutron fluence and chemical composition of material candidates to fulfill the criterion of maximum ΔT_{DBT} (40 °C) to avoid additional safety calculations according to KTA 3203.
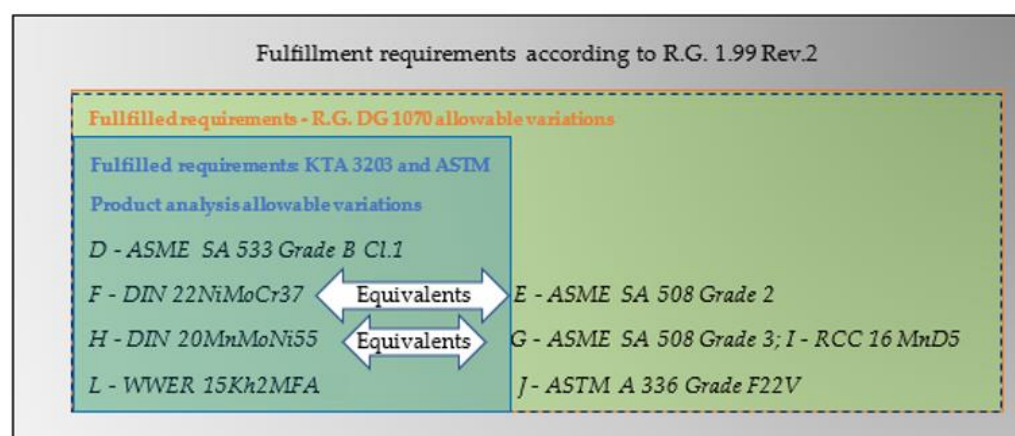
Figure 11. Selection chart based on requirements considered.

The ASME B&PV and KTA codes require the most stringent material requirements. For forgings (DIN 22NiMoCr37 and DIN 20MnMoNi55), KTA better fulfills the requirements established by R.G. 1.99 Rev.2 and KTA 3203 also considers product analysis variation according to ASTM A-788 and R.G. DG 1070 (valid for metallic commercial-grade spare parts). The consolidated SA-533 Gr.1 meets the requirements of R.G. 1.99 Rev.2, KTA 3203, ASTM A-788 and R.G. DG 1070. ASME SA-508 Gr. 2, SA-508 Gr.3, and RCC 16MnD5 fulfill the requirements of R.G. 1.99 Rev.2, KTA 3203 and R.G. DG 1070. As shown in Figure 11, if equivalent grades are compared (DIN 22NiMoCr37 to ASME SA 508 Grade 2, and DIN 20MnMoNi55 to ASME SA 508 Grade 3 and RCC 16MnD5), it is observed that KTA requirements are more stringent. Regardless, in addition to irradiation embrittlement

a stringent control of defects like flaws (generated in manufacturing and/or in service) is crucial. Thus, the main manufacturing and in-service inspection codes define the reference defect sizes (Table 8) from a perspective based on the RPV mechanical integrity and fitness-for-service.

Table 8. Reference defect sizes usually used [85,86].

ASME B&PV XI	KTA	RCC-M	Russian (GOST)
$t > 300$ mm $\rightarrow a/c = 1/3$, $a = 75$ mm 100 mm $> t \leq 300$ mm $\rightarrow a/c = 1/3$, $a = t/4$ $t \leq 100$ mm $\rightarrow a/c = 1/3$, $a = 25$ mm	$a = 1/4t$ $2c = 1.5t$	$a = $ min $(0.25$ t, 20 mm) t > 40 mm $a/2c = 1/6$	$a = t/4$ $a/c = 1/3$

Notes: a = crack depth, c = crack semi-length, and t = wall thickness; considered $t = 245$ mm.

For ASME B&PV, KTA, RCC-M and GOST, geometrical equation is $2c = 6a$. RCC-M is the most stringent since it demands $a = $ min $(0.25t, 20$ mm) for the RPV thickness $(0.25t > 20$ mm). Additionally, ASME in the latest edition (2019) of Section XI of ASME B&PV code has replaced Division 2 [87] and removed Division 3 from Section XI. The 2019 edition has replaced Division 2 with the requirements for Reliability and Integrity Management (RIM) programs for nuclear power plants [88]. This reinforces the idea that nuclear requirements are in continuous development, recently incorporating the best practices from other industries like oil and gas to be used in the Remaining Useful Life (RUL) estimation analyses based on cracks detection.

4. Conclusions

In this work, a novel methodology has been presented enables the analysis—from a historical perspective—the technical heritage made up of several regulations, technical codes and their standardized requirements for the manufacture and in-service inspection of RPVs. In addition, an analysis focused on the differences between irradiation embrittlement prediction models and acceptance criteria for detected defects (both during manufacturing and in-service inspection) as described by the different technical codes required by different national regulations has been performed. The novelty of this new approach lies in the idea of integrating a review, analysis and comparison of some relevant aspects related to materials, manufacturing and in-service inspection included in international codes and specifications based on them. As general considerations regarding the analysis of the manufacturing and in-service codes, the following can be underlined:

- ASME BP&V, KTA, RCC-M or PNAE are reference codes, which in relation to requirements and material tests refer to ASTM, DIN, NF, ISO standards, among others. Therefore, the reading of codes is tedious, and the use is not immediate.
- In order to carry out the selection of materials and the tests that will determine the mechanical properties of the steels used in the manufacture of reactor vessels, it is necessary to consult a large number of publications and studies to establish thresholds for chemical composition. It is recommendable to use prediction models for ductile-to-brittle temperature shift to select the most suitable materials considering the long-term in-service behavior.

Focusing on the quantitative analysis of the standardized materials specifications according to the different regulations, the following conclusions can be highlighted:

- Most of the results from the historical studies presented in this work on the influence of chemical composition, neutron flux and temperature on the materials performance, under reactor operating conditions, are still considered valid today, since they have been confirmed by analyzing the materials from the capsules of surveillance from reactors that have been in operation for decades [19,89–92].
- The most stringent materials requirements for RPV manufacturing are provided by the American and German codes. RCC-M is the most stringent with respect to the reference defect size.

- Regardless to the previously described, it is concluded that although American code remains the "gold standard" for RPV manufacturing and in-service inspection, KTA, RCC-M and GOST standards also provide stringent requirements, and they are recognized codes.

Finally, it is crucial to highlight that inspection programs have been historically developed and implemented based on operational experience in order to ensure the structural integrity of the components during service life, since:

- More accurate and reliable validated NDE systems are available and automated for using in in-service inspections.
- Inspection plans have been increasingly improved to identify and evaluate better the potential materials degradation mechanisms.

In the future, further research related to the analysis of technical documentation, from a historical perspective, for equipment manufacturing and in-service inspection in demanding power generation applications will be developed.

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