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Beyond RPV Design Life

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О возможности превышения расчетного ресурса корпусов атомных реакторов

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Проанализированы стандартные программы мониторинга расчетного ресурса (до 40 лет эксплуатации) корпусов атомных реакторов с использованием образцов-свидетелей. Ввиду усовершенствования методов испытаний и оценки радиационного охрупчивания материалов отмечается необходимость пересмотра действующих программ и проведения научного исследования с целью уточнения ресурса прочности корпусных материалов (продления его до 60 лет). Выполнено прогнозирование долговечности корпусных сталей на основании результатов, полученных на образцах-свидетелях, образцах типа Шарпи, а также с использованием метода "Master Curve".

Ключевые слова: корпусная реакторная сталь, программа наблюдения, расчетный ресурс, прогноз ресурса (долговечности).

Introduction. The highest priority key Category 1 component identified in all separate national categorization exercises has been the reactor pressure vessel (RPV). This is because the RPV is considered irreplaceable or prohibitively expensive to replace. This, in turn, means that if it degrades sufficiently, it could be the operational life-limiting feature of the nuclear power plan (NPP). The RPV houses the reactor core, and because of its function it has direct safety significance.

The majority of the early Westinghouse designed plants had a "design basis life," as distinct from "physical life," of 30–40 years. Specific "design basis life," such as 40 years, was not based on technical studies of material degradation in general, but was based on fatigue usage factors, for the most part. Operation of existing LWRs longer than originally intended has now become a relevant feature. Radiation embrittlement is the most important degradation mechanism limiting the RPV life. Neutron irradiation degrades the mechanical properties of RPV steels, and the extent of the degradation is determined by a number of factors such as neutron fluence, irradiation temperature, neutron flux, and the concentration of deleterious elements in the steel.

As a result of technical and economic considerations, the operating life of an NPP could be easily 50 or 60 years [1]. This might require upgrading of the RPV surveillance programs and the use of modern techniques and approaches such as Charpy reconstitution and Master curve testing to cover the extended operation time.

RPV Surveillance Data. Changes in the material properties due to neutron irradiation are monitored by means of surveillance programs. Every PWR and BWR pressure vessel has an ongoing RPV material radiation surveillance program. To date several hundreds surveillance capsules have been removed from their host RPVs and tested. The results from these surveillance capsules have been used to develop heatup and cooldown curves and to analyze all potential or postulated accident or transient conditions.

The structural integrity evaluation of the RPVs of the Spanish reactors follows the regulations, guidelines, codes, and standards developed in the US, since the reactors were designed by Westinghouse and General Electric, and are similar in design and operation to the American reactors. An exception is the Trillo I reactor supplied by Siemens KWU (now Framatome ANP). Table 1 lists the operating reactors in Spain. José Cabrera reactor will be decommissioned in 2006 after a useful life slightly lower than the design life. Its total gross electrical production will be around 35.000 GWh by the year 2006. It can be observed in Table 1 that, in general, the weld is the most limiting material of the beltline region for the older reactors.

Table 1

Spanish Reactors in Operation

Nuclear power plant	Type of reactor	Supplier	Initiation of operation	Beltline limiting material for brittle fracture	Surveillance capsules analyzed in 04/2003
José Cabrera	PWR	Westinghouse	1968	Weld	4
Almaraz I	PWR	Westinghouse	1981	Base metal	4
Almaraz II	PWR	Westinghouse	1983	Base metal	3
Ascó I	PWR	Westinghouse	1983	Base metal	3
Ascó II	PWR	Westinghouse	1985	Base metal	3
Vandellós II	PWR	Westinghouse	1988	Weld	3
Trillo I	PWR	Siemens-KWU	1988	Weld	1
Santa Ma de Garoña	BWR	General Electric	1971	Weld	3
Cofrentes	BWR	General Electric	1984	Weld	2

It is well known that certain residual elements, such as copper and phosphorus, favor embrittlement. With a content of more than 1%, nickel also causes embrittlement, although it has a beneficial effect since it produces a lower initial value of the reference temperature RT_{NDT}. The role played by other residual elements such as tin, antimony, and arsenic is not clear. Copper produces fine precipitates, which cause embrittlement by making the movement of dislocations more difficult. Copper and nickel are believed to have a synergy effect of hardening. Phosphorus embrittlement is a result of two mechanisms. On the one hand, fine precipitates are formed in a manner analogous to the case of copper, and on the other one, phosphorus precipitation is segregated at the grain boundaries causing them to be weakened (non-hardening embrittlement) and leading to the possibility of unstable crack growth. Table 2 shows the range of concentration of Cu, Ni, and P measured in the RPV surveillance steels of the Spanish LWR reactors.

Table 2

Concentration of Deleterious Elements in Steel

Material	Range of Cu (%)	Range of Ni (%)	Range of P (%)
Base metal	0.03–0.14	0.50–0.77	0.005–0.013
Weld	0.02–0.30	0.04–1.01	0.004–0.015

Because phosphorus can also contribute to hardening, it is not always clear if its effect on the embrittlement is due to hardening or segregation, or both. A predominant role for hardening would be consistent with the observed characteristically low susceptibility of the Spanish PWR vessel steels to the embrittlement phenomena caused by grain boundary segregation of impurities.

The ratio of the yield stress change to the increase in $T_{41\,\mathrm{J}}$ measured by Charpy testing is frequently a good measure of whether non-hardening embrittlement occurs. Figure 1 shows this ratio for the Spanish PWR vessel steels. The scatter observed at low fluence values is not relevant since small increases in the yield strength can produce high $\Delta T_{41\,\mathrm{J}}/\Delta y_s$ ratios. The conclusion from Figure 1 is that there are no non-hardening embrittlement components in the Spanish PV steels under consideration.

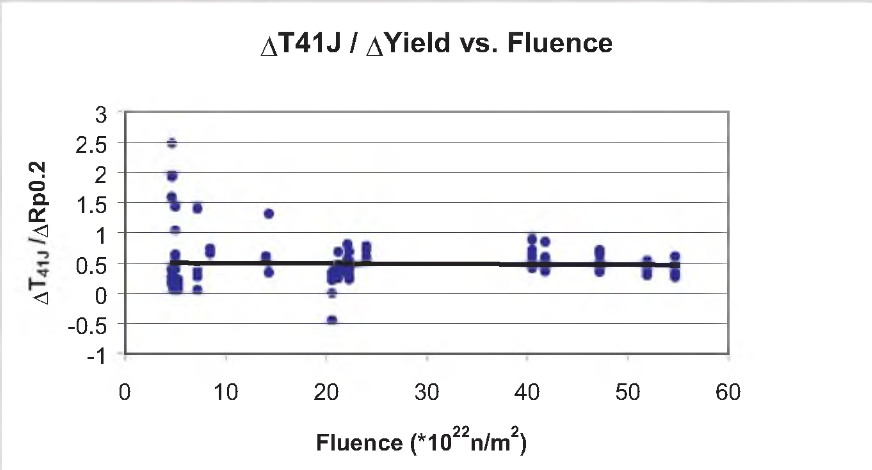

Fig. 1. The absence of non-hardening embrittlement in the Spanish PWR PV steels.

Until now, a total of 26 surveillance capsules have been tested and analyzed in Spain. The surveillance data allow one to verify the theoretical embrittlement trend curve and to detect any anomaly in the irradiation conditions. Figure 2 shows a comparison of measured and predicted values using the Eason model [2]. The predictions are reasonably good at low fluence levels, but become more dispersed at higher fluence levels.

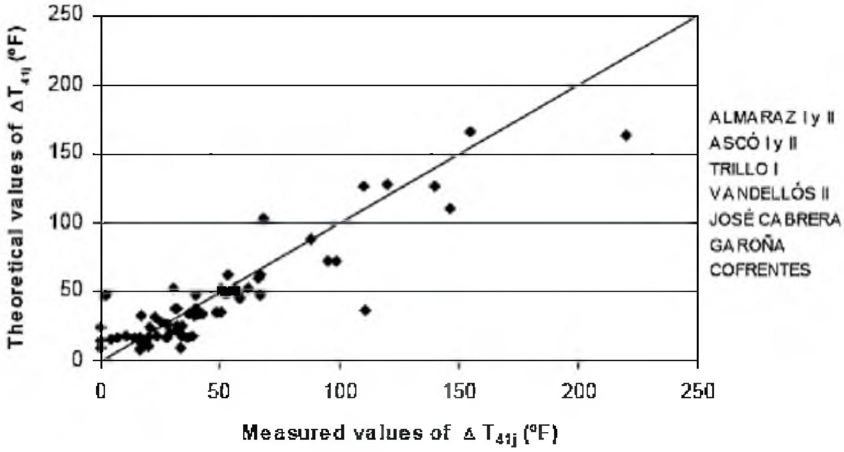

Fig. 2. Comparison of measured and predicted shifts using Eason correlation.

Prediction to 40 Years. The steels used for reactor vessels become progressively brittle throughout the service life of the component as a result of the effect of neutron irradiation to which they are exposed. This progressive degradation must be known in order to guarantee the structural integrity of the vessel throughout its service life. There are two key parameters, which make it possible to evaluate and quantify the vessel degradation. These parameters are the *USE* (Upper Shelf Energy) and RT_{NDT} (Reference Temperature). The initial values of the RT_{NDT} are obtained by means of Charpy and drop-weight tests, while increases in the RT_{NDT} and the value of *USE* are measured exclusively by means of Charpy tests. The RT_{NDT} tends to increase throughout the service life of the reactor, while the *USE* tends to decrease.

The American Code for a pressurized thermal shock (PTS) 10CFR50.61 [3] establishes requirements on the ability of the reactor vessel in pressurized water reactors to withstand events wherein the vessel is both rapidly overcooled (thermally shocked) and pressurized (or repressurized). The Code requires calculation of the projected values of the RT_{NDT} at the end of life (EOL) and comparison with given limit values. The PTS screening criterion is 270°F for plates, forging and axial weld materials, and 300°F for circumferential weld materials.

On the other hand, Regulatory Guide 1.99 revision 2 [4] requires for RPV beltline materials of new plants that the reference temperature RT_{NDT} at the $1/4T$ position in the vessel wall at the end of life be less than 200°F. This could be also considered a good recommendation for existing nuclear power plants.

Table 3 shows the projected values of the RT_{NDT} at 32 *EFPY* (effective full power years) for the most limiting beltline material of the Spanish RPVs at the inside location of the reactor vessel wall. In the past, 32 *EFPY* were associated with 40 calendar years of operation, representing an average availability factor of 80%. For PWRs operating in Spain, the projected RT_{NDT} values are lower than the PTS limit values established in 10CFR50.61. Moreover, all the RT_{NDT} values for the PWRs and BWRs listed in Table 3 are below the recommended value of 200°F.

Table 3

Values of RT_{NDT} and *USE* after 32 *EFPY*

RPV	RT_{NDT} (°F)	*USE* (J)
No. 1	146	71
No. 2	138	*USE* > 104
No. 3	156	*USE* > 68
No. 4	114	79
No. 5	107	95
No. 6	115	*USE* > 123
No. 7	106	163
No. 8	119	104
No. 9	90	114

As established in Appendix G of 10CFR50, initially reactor vessel beltline materials must have Charpy upper shelf energy, *USE*, in the transverse direction for the base material and along the weld for the weld material of no less than 102 J, and must maintain the USE of no less than 68 J throughout the life of the vessel, unless it is demonstrated that lower values of the *USE* can provide margins of safety against fracture equivalent to those required by the ASME Code. Regulatory Guide 1.161 [5] was developed by the US NRC to provide a comprehensive guidance for evaluating RPVs when the Charpy upper shelf energy falls below the 68 J limit of Appendix G to 10CFR50.

All the *USE* values at 32 *EFPY* listed in Table 3 are over the 68 J limit value. In several cases it was difficult to get an accurate projected value of the upper shelf energy at 32 *EFPY*, since the Charpy-V impact tests of unirradiated material were performed at the temperature in the upper shelf region not high enough. For these cases, the projected *USE* values were based on the surveillance data of capsules, which accumulate higher neutron fluence than the projected one inside the reactor vessel wall at 32 *EFPY*.

It is worth to mention two specific types of activities performed in Spain, namely, analysis of the impact of power uprating on the RPV structural integrity assessment and implementation of ASME Code Cases N-640 [6] and N-588 [7].

Several nuclear power reactors in Spain have undertaken a major initiative to increase the economic value of their plants by increasing the license power at which the plant is permitted to operate. The Spanish nuclear regulatory body, CSN, has reviewed applications for power uprates in Almaraz, Ascó, Vandellós, and Cofrentes reactors in order to determine if adequate safety margins exist at the increased power and to ensure that the regulatory limits are not exceeded. The power uprating studies performed by the utilities include re-evaluation of the projected neutron fluence at the end of life and, consequently, the projected RT_{NDT} and *USE* values and the impact in the pressure–temperature limit curves.

At the Cofrentes nuclear power plant they made use of the ASME Code Cases N-588 and N-640 to update the pressure-temperature limit curves. ASME Code Case N-588 allows the use of an alternative procedure to calculate the applied stress intensity factors of Appendix G of ASME XI for axial and circumferential welds. The ASME Code Case N-640 allows the use of K_{Ic} rather than K_{Ia} to determine the pressure–temperature limit curves. The use of K_{Ic} eliminates one of the conservatisms used to generate these curves.

Towards 60 Years of Operation. Section XI.M31 Reactor Vessel Surveillance of NUREG 1801 (the GALL report) [8] contains recommended actions and acceptable methods to evaluate the embrittlement status of the vessel for a period of until 60 years. For instance, if a plant has a surveillance program that consists of capsules with a projected fluence of less than the 60-year fluence at the end of 40 years, at least one capsule is to remain in the reactor vessel, and it should be tested during the period of extended operation. This is of application to the Spanish BWRs since the lead factor is close to one.

One important remark in NUREG 1801 is the recommendation to remove the standby capsules if the lead factors are relatively high. For example, in a reactor with a lead factor of three, after 20 years, the capsule test specimens would have received neutron exposure equivalent to what the reactor vessel would see in 60 years. Thus, the capsule is to be removed and placed in storage since further exposure would not provide meaningful metallurgical data. The standby capsules would be available for reinsertion into the reactor if additional license renewals are sought (e.g., 80 years of operation). Although at present, the Spanish NPPs are not contemplating any life extension beyond the 40-year term, Almaraz NPP will be the first Spanish plant to follow this NUREG recommendation. Two standby capsules from each unit will be removed from the vessel in the next outage since by that time the standby capsules will have accumulated the neutron fluence slightly higher than the projected one for the vessel at 60 years.

When all the surveillance capsules have been removed, some means must be established to ensure that the on-going exposure of the reactor vessel is consistent with the basis used to predict the effects of embrittlement to the end of life. It is not possible to say now what the operating philosophies will be in the future. The general recommendation is that, if possible, ex-vessel neutron dosimeter be installed one or more fuel cycles prior to withdrawing the last (typically the 60-year) surveillance capsule. The dosimeter is then removed and analyzed simultaneously with the surveillance capsule, and a replacement ex-vessel dosimeter is installed. Simultaneous measurements inside and outside the reactor vessel provide a larger amount of information to characterize the reactor vessel exposure.

Facing a 60-year operation, the most promising techniques are the reconstitution of surveillance specimens and Master curve testing. The former allows us to solve the limitation in the amount of surveillance material available for irradiation and testing, and the latter provides a technically sound approach for defining a unique fracture toughness transition temperature T_0 for ferritic steels clearly superior to the old RT_{NDT}. The potential application of the Master curve approach to the Spanish reactors and the benefits of its use were outlined in [9].

In January 2002, the regulatory body CSN and the utilities represented by UNESA started a 3-year project (CUPRIVA project) focused on the CT and Charpy specimen reconstitution and Master curve testing. Two pilot plants are participating in the project, Santa Maria de Garona (BWR) and Ascó II (PWR), which provided surveillance material for investigation. The available irradiated broken specimens from Ascó II base material are used to machine reconstituted precracked Charpy V-notch (PC-CVN) specimens. The Master-curve testing results of the Ascó II specimens will be compared with fracture toughness results obtained by conventional testing of available irradiated compact test 1/2T CT specimens. The base and weld surveillance materials of the Garona RPV are investigated. Reconstituted CT and PC-CVN specimens are tested according to the Master curve approach. For the Garona reactor it will be possible to compare the results of the Master curve testing of unirradiated and irradiated specimens. Preliminary results will be available by the end of the year 2003. The test results from Asco and Garona reactors will be used to determine the remaining lives of these reactors with a higher accuracy than previous evaluations. The possible application of the ASME Code Cases N-629 and N-631 will be considered. The study will include a comparison of the RT_{NDT} and RT_{T_0} values. The latter indexing temperature, as defined in the ASME Code Cases, is based on the Master Curve concept, and assures that the $K_{\mathrm{I}c}$ curve will bound the actual material fracture toughness data by a functional equivalency of RT_{NDT} and RT_{T_0}.

As it is usual for the BWRs designed by the General Electric, the surveillance practice of the two Spanish BWRs, Cofrentes and Garona reactors, includes reinsertion in the vessel of capsules with reconstituted specimens. The surveillance data provided by these non-mandatory capsules allows a better definition of the embrittlement trend curves for these reactors, and a better prediction of the projected RT_{NDT} and *USE* values at the end of life. In addition to the reinsertion of surveillance capsules, Garoña NPP carried out a modification of the surveillance holder to improve the lead factor of the surveillance capsules that is now slightly higher than one. For Cofrentes NPP, the last capsule, which is being manufactured and will be reinserted in the vessel in October 2003, will include, in addition to the limiting beltline base and weld materials, a reference steel or correlation monitor material, JRQ. This will allow us to detect and analyze any anomaly or change in the irradiation conditions.

Conclusions. The reactor pressure vessel surveillance program is a key factor in the life assessment of the Spanish nuclear reactors, since radiation embrittlement of the RPV is the most life-limiting degradation mechanism, and the RPV is the most important pressure boundary component of the nuclear power plant. Similar design of the Spanish pressure vessels and their surveillance programs are a favorable factor for the evaluation of the surveillance data, which allows an easy identification of anomalies in any specific reactor.

The new embrittlement trend curves developed in the US have been applied to Spanish reactors. These new correlations take into account new independent variables like irradiation temperature, irradiation time in addition to the variables taken into account in previous models (e.g., in Regulatory Guide 1.99, Rev. 2). The results indicate that there is a good agreement between the theoretical values (obtained using the Eason correlations) and the experimental values reported after analyzing the surveillance capsules.

The projected values of RT_{NDT} and *USE* of the beltline limiting materials at 32 *EFPY* show that the pressure vessels could easily operate beyond the 40-year design life. These key parameters were determined using the methodology described in Regulatory Guide 1.99, Rev. 2, which is in force in Spain. Although at present the Spanish NPPs are not contemplating any life extension beyond the 40-year term, the requirements established in NUREG-1801 are taken into account. The main research activities in Spain are dealing with the surveillance specimen reconstitution and Master curve testing.

Резюме

Проаналізовано стандартні програми моніторинга розрахункового ресурсу (до 40 років експлуатації) корпусів атомних реакторів із використанням зразків-свідків. Удосконалення методів випробувань та оцінки радіаційного окрихчування матеріалів потребує перегляду діючих норм і проведення наукового дослідження з метою уточнення ресурсу корпусних матеріалів (подовження його до 60 років). Виконано прогнозування довговічності корпусних сталей на основі результатів, що отримано на зразках-свідках та на зразках типу Шарпі, а також із використанням методу "Master Curve".

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