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SOME ASPECTS OF REACTOR PRESSURE VESSEL INTEGRITY

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1.0 Pressure Vessel Integrity.

The pressure vessel integrity has been a matter of extreme interest since the very advent of reactors. A lot of research studies have been done to resolve the concern that there is no backup to the vessel nor any specific mitigating system to cope with disruptive failure. The vessel failure should be low enough in probability that it need not be considered as a design-basis accident.
Various regulations and regulatory guides have emphasized the importance of operating the vessel within prescribed guidelines. Technical specifications, specially Limiting Conditions for Operation (LCO) are quite specific in determination of acceptable operating regions.

Westinghouse analysis of reactor vessels is performed by ASME Code XI, Section III Appendix G for normal, upset and test conditions. The analysis methods are applied to four locations in the reactor vessel: closure head to flange region, nozzle to shell course region, beltline region and the bottom closure head to shell course region.
The reactor beltline is defined as that area which is adjacent to the reactor core and is of interest because of neutron flux in its area. Neutron fluence (time integrated neutron flux) is used in the calculation of exposure effects and changes in Nil-Ductility Reference Temperature (RT_{NDT}). As the reactor vessel ages , the shift in RT_{NDT} in the beltline region becomes the most restrictive limit. The reactor vessel material becomes brittle at higher temperatures than unirradiated (new) material.

Since the reactor vessel is not uniform in its construction, several different values for reference temperature of RT_{NDT} and stress intensity factors must be calculated and used to generate the limit curves. The determination of RT_{NDT} involves the testing of reactor vessel material samples.

1.1 Pressure Vessel Safety Research.

Pressure vessel safety research results provide bases for regulatory positions, specially on following topics:

- fracture analysis methods for determination of Pressure-Temperature (P-T) limits
- developing embrittelment estimation methods
- identifying significance of copper in irradiation damage
- developing and validation of Pressure Thermal Shock (PTS) analysis methods
- demonstrating effectiveness of thermal annealing

1.2 On-going activities.

The most active on pressure vessel safety research is nuclear industry worldwide and specially in United States research institutions and regulatory bodies.
Currently on-going activities in United Stations are oriented to amend rules of Code of Federal Regulations (CFR):

- 10 CFR 50.61 : PTS rule
- 10 CFR 50, App. G : Fracture Toughness Requirements
- 10 CFR 50, App. H : Surveillance

US Nuclear Regulatory Commission (US NRC) has made a general conclusions about pressure vessel safety research:

- the results of research has contributed to safe regulation of pressure vessels
- recent experiences are good bases for further work
- planned research on reactor vessel stresses problem resolution
- designed program to anticipate plant aging problems
- close interaction with nuclear industry and research institutions

2.0 Pressure versus Temperature limits.

Generally, in Pressure-Temperature diagram for PWR reactor coolant system there are three principal challenge paths.

Path A represents a typical design basis pressurization transient such as load reduction. Path B is a generalization of PTS scenario and path C is a Low Temperature Overpressure Protection (LTOP).

Ideally the licensee would estimate the likelihood of occurance of these three paths and probability of the vessel conditional failure.
In a deterministic sense, for example, US NRC expect that the reactor coolant pressure boundary behaves in a non-brittle manner. The probability of rapidly propagating fracture must be minimized, as it is required by General Design Criteria (GDC) No. 31.

2.1 Limiting Conditions for Operation.

Standard format of plant Technical Specifications includes in section 3.4 Limiting Conditions for Operation (LCO 3.4.9.1). The most important limiting conditions for Nuclear Power Plant Krško (3) are:

- a maximum heat up rate 27.7 ^{0} C in any 1-hour period
- a maximum cooldown rate 55.6 ^{0} C in any 1-hour period

- a maximum temperature change of less than or equal to 5.6 ^{0} C in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves

The limit curves for Krško Nuclear Power Plant for heat up and cooldown are calculated for 15 Effective Full Power Years. Determination of LCOs reqires calculation of nonstationary stress and temperature fields in irradiated region of reactor vessel. It is necessary to perform mechanical tests on irradiated samples of reactor vessel material to determine the temperature of brittle fracture and other parameters necessary for accurate calculations.

3.0 Relocation of P-T limit curves from plant Technical Specifications.

US NRC has issued on January 31,1996 a Generic Letter 96-03 " Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits".

During the development of the improved standard technical specifications a change was proposed by nuclear industry to relocate the P-T curves and LTOP setpoint curves and values currently contained in the TS to a licensee-controlled document.
The licensee must be able to reference a methodology for developing the curves and setpoints containing in Pressure Temperatures Limiting Report (PTLR) and also the methodology used in process of developing such report.
Relocation of the curves and setpoints to a licensee controlled document requires some separate licensee actions, like:

- developing the methodology for relocation approved by regulatory body

- developing the PTLR report which contains all necessary data with explanation

- modify the applicable sections of the TS accordingly

Detailed above mentioned requirements are described more precisely in GL 96-03, Attachment 1.

4.0 Vessel failure probability.

In the year 1981 was issued an assessment of the failure of reactor vessels in NUREG-0778. The purpose of this study was to include better definition of the joint effects of radiation, materials and flaw distribution and specially current limitations on fracture mechanics at that time. The report estimates the failure rate for normal startup and shutdown sequences to be in the range of 1 E-6 per vessel-year.

Dependent on flaw distribution and at end-of-life fluence the likelihood of failure for a pressure of 20.68 MPa (3000 psi) was estimated to 1 E-4. As the conclusion a suggestion was made that it would be appropriate to compare failure probabilities of nonnal and transient operation with startup and shutdown conditions and sequences.

Further research work in US mainly deals with vessel failure prompted by seismic events, low temperature overpressurization or pressure thermal shock (PTS) scenarios.

The reactor safety study (WASH 1440) defined a reactor vessel rupture event as a rupture large enough to negate the effectiveness of the Emergency Core Cooling System (ECCS). The median value of the probability was stated to be 1 E-6 per vessel-year and as such is negligible contributor to PWR risk.

US NRC observed, for example, the properties of the Yankee Rowe reactor vessel and thought that the probability of pressure thermal shock failure should be kept below the 1 E-4 per year, using best estimate parameters. However, risk assessments in general do not model the vessel failure as an initiator.

5.0 Conclusions.

For PWR reactors, there are practically two safety limits during normal operation and transients: core safety limit (fuel cladding heat transfer) and primary system boundary pressure safety limit. Both these limits contains important process variables that are found to be necessary to reasonably protect the integrity o certain of the physical barriers that guard against the uncontrolled release of radioactivity.

A typical vessel safety limit on pressure might be on the order of 10% in excess of the design pressure. The Technical Specifications contain Limiting Conditions of Operation (LCO) on pressure-temperature combinations. The concern is to provide a margin to brittle failure of the reactor vessel. Violating LCO limits might result in a consequent nonisolable leak or loss of coolant accident. The most important parameters to be observed are pressure, temperature and heat up and cooldown rates of primary coolant. In all operational modes the nuclear power plant must operate within specified pressure-temperature limits.

References:

1. NRC Generic Letter 96-03: Relocation of the Pressure Temperature Limit curves and Low Overpressure Protection System Limits, January 31, 1996

2. 10CFR 50.61, PTS Rule,10CFR 50, App. G and H (Fracture Toughness Requirements, Surveillance Requirements)

3. NEK Technical Specifications, rev. 45, 18.12.1995