LAGUNA VERDE

Reactor Pressure Vessel Neutron Embrittlement & Metal Fatigue

COMISIÓN FEDERAL DE ELECTRICIDAD

SUMMARY

Laguna Verde NPP, has two units, with capacity 682.44 net MWe each. Both operating licenses were granted for a period of 30 years, even if by design (analysis and calculations) the main components have a life of 40 years. Hence the expiration of these licenses will be in 2020 for Unit 1 and 2025 for Unit 2.

For several years the Nuclear Power Plants in the United States, began working on a process called License Renewal. Now Laguna Verde has started this process and one of the main systems to evaluate is Vessel and its internals.

This report summarize the work done and scheduled of investigation and evaluation of design documents, operational experience (external and internal), and provide a preliminary base line for the following evaluation work. It has been reviewed information of BWRVIP, NUREG 1800 & 1801; 10CFR50 G & H and OIEA regarding vessels analysis, based on this information we defined the subcomponents which would be analyzed

Also has been established a preliminary subcomponent-age mechanisms and what information would be necessary to review in order to confirm the age mechanism, determine the actual subcomponent condition, which mitigation or preventive activities are taken and which would be necessary to plan and best moment for doing. A brief revision of ISP (Integrated Surveillance Program) of BWRVIP also was made and of the ongoing Plant Surveillance Program. Regarding the surveillance of beltline materials, program based on appendixes G and H of 10CFR50. Also we have considering our participation on this Integrated Surveillance Program.

GENERAL DESCRIPTION

Laguna Verde power station is located on the coast of the Gulf of Mexico, has two units BWR-5, supplied by General Electric Company with the following features:

Units	2 X 682.44 net MWe/ 815 MWe after EPU
Type	Boiling Water Reactor (BWR-5)
Control	109 control rods of SS with Boron carbide or Boron carbide plus Hafnium
Pressure	70.69 Kg/cm^2 nominal.
Power (thermal)	2027 MWt before EPU/ 2317 MWt after EPU
Steam flow	3961.3 Ton/h at 99.7%/ 4536.36 Ton/h at new 100%
Recirculation pumps	2
Recirc. Nominal flow	9235 Ton/h (at 100%)
Feed water flow	3950 Ton/h/ 4450 Ton/h for EPU
Jet pumps	20
Vessel	Made of carbon steel, internally clad with SS,
	Total height; 20.8 m, diameter 5.30 m and
	Thickness from 13 to 18 cm.

The reactor vessel is a vertical, cylindrical pressure vessel of welded construction. The vessels for Laguna Verde were designed, fabricated, tested, inspected and stamped in accordance with ASME code section III class A included the Winter addenda of 1971. Design of the reactor vessel and its support system meets Seismic Category I. The cylindrical shell and bottom head sections are fabricated of low alloy steel, it was internally protected with stainless steel weld overlay. Nozzle and Nozzle weld zones are unclad except for those mating to stainless steel piping systems.

Materials & Fabrication

The reactor pressure vessel was primarily constructed from low alloy, high strength steel plate and forgings. Plates were ordered of SA 533 grade B, class1 and forgings were SA 508 class2. These materials were melted to fine grain practice and supplied in quenched and tempered condition. Studs, nuts and washers for the main closure flange were made of SA 540 grade B23 or B24. Welding electrodes were low hydrogen type as SFA 5.5

All plates, forgings and bolting were ultrasonically tested and surface examined by magnetic particle or penetrant liquid methods. Fracture toughness properties also were measured.

Submerged arc and manual electrode welding processes were employed. Electro slag welding was not permitted. Preheat and inter pass temperatures for welding of low alloy steel, meet or exceed the requirements of ASME III section, post weld heat treatment at 1100°F was applied to all low alloy steel welds. Radiographic examination was performed on all pressure containing welds, in addition also was performed supplemental ultrasonic examination to these welds

Service conditions:

Operating power	2317 MWt (120 % of nuclear boiler rated power)
Vessel dome pressure	≤ 1020 psig
Steam flow	4536.36 Ton/h (100% of nuclear boiler rated steam flow, 2317 MWt.)

EVALUATION FOR LONG TERM OPERATION:

As a result of license renewal evaluation all components of vessel and vessel internals within the scope are going to be evaluated, considering materials, environments and aging mechanisms and it has to be demonstrated that the effects of aging are adequately managed, so that the intended functions will be maintained consistent whit the current licensing basis.

The Vessel and internals structures and components will be evaluated for neutron embrittlement, specifically the belt line region for vessel and core shroud. Neutron fluence evaluation, USE (Upper shelf Energy), RTNDT (Reference Temperature for Nil Ductility Transition and ART (adjust reference temperature) are activities of the actual RPV Surveillance program for non-ductile failure under 10CFR50 Appendix G, and for extended period purposes will be projected to 54 EFPY (effective full power years).

REACTOR PRESSURE VESSEL NEUTRON EMBRITTLEMENT

Irradiation by neutrons results in embrittlement of carbon and low-alloy steels. It may produce changes in mechanical properties by increasing tensile and yield strengths with a corresponding decrease in fracture toughness and ductility. The extent of embrittlement depends on neutron fluence, temperature, and trace material chemistry. Neutron embrittlement is a key element to be addressed for plant life, through Vessel Material Surveillance Program and it will be maintained for license renewal period.

The surveillance program for Laguna Verde was established based on 10CFR50 appendix G and H requirements to assure the brittle fracture of reactor vessel is prevented. It consists for each unit, of three surveillance capsules and one separate flux dosimeter. Each capsule contains Charpy specimens of the beltline base, weld and HAZ materials and a set of flux wires used to determine the fluence experienced by the capsule. The surveillance capsules are scheduled to be withdrawn periodically during the plant life (current schedule is what recommended by ASTM E-185-82, a capsule at 6, 15 and 32 effective full power years). In addition to the flux wires in the surveillance capsules, a flux wire dosimeter is attached to the capsule at 30° for removal after the first fuel cycle. Since the vessel fluence is proportional to the thermal power produced, the results of the flux wire dosimeter test are used to provide a calibration point of vessel fluence versus accumulated thermal power.

Since the first capsule was withdrawn each Laguna Verde unit, we took the option to reconstitute de surveillance specimens and reintroduce one cycle after, the objective to perform this work was to prepare for the potential operating extended period, so now U1 has a complete set of surveillance capsules (three), one of them from the original program and two reconstituted, and U2 has two remaining capsules, one original and one reconstituted, having with this enough surveillance material for 20 additional years.

The surveillance material was reconstituted as follows; charpy specimen pieces after the testing were cut for removing the deformed surface, each side was welded with an insert by stud-weld process and finally prepared as a mini charpy specimen. A mini tension specimens were prepared from the resultant pieces of the originals tested. A new capsule was fabricated and prepared with the specimens reconstituted for being reinstalled during the next outage from its withdrawn.

Regarding the work on going in preparation for License Renewal, following items are proposed and evaluated regarding the effect of neutron embrittlement on the RPV beltline materials, reactor pressure vessel, internals, and reactor internals repair hardware and aging management for the period of extended operation (PEO).

- Neutron Fluence Projections
- Upper-Shelf Energy I Adjusted Reference Temperature
- RPV Reflood Thermal Shock (Service level D fracture mechanics evaluation)
- Axial, Circumferential , and Nozzle to Shell Weld Inspection
- RPV Flaw Evaluation Revision
- Reactor Internals Repair Hardware loss of Preload
- RPV Surveillance Program Update

➢ <u>Neutron Fluence Projections:</u>

- Plant-unique analyses is being performed to determine reactor pressure vessel (RPV) and component fast neutron fluences (energy >1 .0 MeV) for critical materials in the RPV beltline region. Following are some important features of the proposed analyses.

- A plant-unique, three-dimensional model shall be used in RPV and component fluence analyses.

- The geometry model for the plant are based upon as-built parameters, if available, that accurately represent the three-dimensional form and positioning of the RPV, vessel internals, and reactor core region.

- The irradiation period for determining fluence in the plant are based on detailed reactor operating history data that includes core designs, fuel designs, core flow data, daily power levels, and projected operating strategies

- Peak fast neutron fluence and location in the RPV and internal components are reported at the end of operating cycle (EOC) 15 for Unit 1 and EOC 12 for Unit 2 and at the end of the extended operating license life (i.e ., 54 effective full power years (EFPY)) for both units.

- EPRI's RAMA Fluence Methodology (RAMA) software are used to determine the best estimate fast neutron fluence for LVNPP1 and LVNPP2's RPV and internal components

➢ <u>Upper Shelf Energy (USE), RTNDT and ART Evaluation:</u>

Neutron embrittlement raises the initial nil ductility reference temperature, RT_{NDT}, and lowers the Upper Shelf Energy (USE) of ferritic materials. This consequence of this increase in RT_{NDT} is the requirement for a higher metal temperature in order for the material to retain the same margin against non-ductile failure as existed in the un-irradiated state. The adjusted reference temperature (ART) is defined as $RT_{NDT} + \Delta RT_{NDT}$ + margin, as defined in the NRC Regulatory Guide (RG) 1.99, Rev. 2;

For this task will calculate the ΔRT_{NDT}, ART, and USE, for all belt line materials, in accordance with NRC RG 1.99, Rev. 2 for each RPV material exposed to a cumulative fluence greater than $1.0x10^{17}$ n/cm^2 at each of two different effective full power years (EFPY). (32 and 54 EFPY). The ART and USE calculations will use the results of the fluence calculations and the RPV material information compiled.

➢ <u>Evaluation of Service Level C/D Events for Non-Ductile Failure (RPV Thermal Shock)</u>

A generic fracture mechanics evaluation was performed in 1979, based upon BWR vessel material properties and cumulative fluence for 40 years of operation, to evaluate the effects of a postulated Loss of Coolant Accident (LOCA) on the structural integrity of a BWR pressure vessel. The original analysis showed that the material fracture toughness at that point in the event exceeded the applied stress intensity factor by a significant margin, and therefore an existing flaw in the vessel would not propagate due to brittle fracture during a LOCA. An updated 60-year fracture mechanics evaluation will be performed for the reflood thermal shock event using plant-specific reactor pressure vessel data for CNLV Unit 1 and Unit 2.

- RPV Vertical (Axial) Weld and Nozzle to Shell (Circumferential) Weld Examination Relief

CNLV performed a plant-specific analysis for the U1 and U2 RPV vertical weld and nozzle to shell weld examination relief, and has changed the scope of in service inspections using the ASME code case N-702, through the application of BWRVIP-108 "Technical Basis for the Reduction of Inspection Requirements for Boiling Water Reactor Nozzle to Vessel shell welds and nozzle blend radii". For RPV vertical welds, the U. S. plants are expected to achieve 'essentially 100 %' inspection coverage. In the event that the actual inspection coverage does not satisfy the 'essentially 100%' inspection requirements, a plant-specific analysis will be conducted to show the component reliability for the actual inspection coverage. Is use the Vessel Inspection Program Evaluation for Reliability Program (VIPER) and methods consistent with applicable BWRVIP guidance, and applicable ASME Code Cases [h], as appropriate. If plant-specific geometry requires development of plant-specific finite element models, then CNLV will build and analyze the plant-specific models. A project-specific V&V software verification will be performed for the VIPERNOZ program and will be documented. A bounding analysis will be performed for both CLV units.

- RPV Flaw Evaluation Revision:

CNLV perform all analysis necessary to update/revise existing RPV flaw evaluations to consider updated fluence evaluations representative of 54 EFPY. CNLV will develop necessary analytical models, including finite element models, calculate or tabulate necessary loads and load combinations, and evaluate all existing flaws against applicable acceptance criteria. CNLV will use industry accepted methods for all structural analysis and will remain consistent with NRC accepted methodologies where available.

- Reactor Repair Hardware Evaluation:

CNLV evaluate the existing reactor repair hardware (Jet pump, core spray, etc.) to ensure that all specified fastener and clamping device preloads will remain within specification considering end of license fluences.

- RPV Surveillance Program Update:

The surveillance program for Laguna Verde was established based on 10CFR50 appendix G and H requirements as was mentioned before, and it will be updated the withdrawn schedule considering the operating extended period to assure the brittle fracture of reactor vessel is prevented.

METAL FATIGUE

The other key element for Vessel and its internals for long term operation is Fatigue, which is a phenomenon leading to fracture under repeated or fluctuating stresses having a maximum value less than the tensile strength of the material. Fatigue fractures are progressive, and grow under the action of the fluctuating stress. Fatigue due to vibratory and cyclic thermal loads is defined as the structural degradation that can occur from repeated stress/strain cycles caused by fluctuating loads (e.g., from vibratory loads) and temperatures, giving rise to thermal loads. After repeated cyclic loading of sufficient magnitude, microstructural damage may accumulate, leading to macroscopic crack initiation at the most vulnerable regions. Subsequent mechanical or thermal cyclic loading may lead to growth of the initiated crack. Vibration may result in component cyclic fatigue, as well as in cutting, wear, and abrasion, if left unabated. Vibration is generally induced by external equipment operation. It may also result from flow resonance or movement of pumps or valves in fluid systems. Crack initiation and growth resistance is governed by factors including stress range, mean stress, loading frequency, surface condition, and the presence of deleterious chemical species.

The Fatigue evaluation will be constituted for:

- A Metal Fatigue Scoping and Recommendation study, which establishes the primary structure for the Fatigue Management Program (FMP). CNLV will produce two documents: a general recommendation report for developing a comprehensive Fatigue Management Program (FMP) and an Environmentally-Assisted Fatigue (EAF) screening analysis. Also it will be updated the actual fatigue calculations.

- Evaluation of Plant Cycle Counting and Fatigue Monitoring Locations:

  CNLV will perform a comprehensive review of existing plant-specific systems, cycle counting procedures, and requirements at CLV. Cycles will be projected for 60 years of operation for each unit.

- EAF Screening:

As part of the closure of GSI 190, the NRC concluded that the utilities requesting license renewal must consider the management of EAF in their aging management programs. Based on this requirement, all plants involved in the license renewal process must address plant-specific evaluation of reactor water environments. CFE will perform an EAF screening analysis using the rules of NUREG/CR-6909 to determine the sentinel locations for each component, which will include those locations specified by NUREG/CR-6260.

CONCLUSION

All the work described here is being performed to evaluate and justify the extended operation period, the analysis and calculations existing in the current licensing basis were performed based on a 40 year life, however the industry experience has demonstrated that it´s possible to follow operating beyond 40 years, Laguna Verde expects to obtain an extended licensed period for 20 years more, and we are aware that as a result of this evaluation CNLV will has to implement the aging management programs determined to manage aging effects for the vessel an its internals components, along with the demonstration that the identified aging effects will be adequately managed, so that there is reasonable assurance that the intended functions will be maintained consistent with the current licensing basis.

REFERENCES:

[1] 10CFR54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants”.

[2] RG 1.188 “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses” July 2001

[3] NUREG 1800 R2 “Standard Review Plan for License renewal Applications for Nuclear Power Plants” USNRC, December 2010

[4] NUREG 1801 R2 “Generic Aging Lessons Learned (GALL) Report”, USNRC, December 2010.

[5] NEI 95-10 R6 “Industry Guideline for Implementing the Requirements of 10CFR54 – The License Renewal Rule” Nuclear Energy Institute, June 2005.

[6] 10 CFR 50 Appendix G and H

[7] FSAR/ISSE de la CNLV.

[8] Especificaciones Técnicas de Operación Unidades 1 y 2 de la CNLV.