PERIODIC SAFETY REVIEWS IN BELGIUM

P. Govaerts, AIB-Vinçotte Nucléaire.

1. THE LICENSING BASIS

Periodic Safety Reviews (PSRs) are requested by the operating licence of each Belgian nuclear power plant.

It is required that 10, 20, 30,... years after the plant has reached its nominal power, the Utility and the Safety Authority jointly proceed to a comparative examination of the design, construction, operating rules and procedures of the existing plant with respect to the current rules and practices in use in the USA and in the European Community at the time of the review.

As the technical support to the Competent Authorities, supervising nuclear safety, AV Nuclear is in charge of that examination and acts as the Regulatory Body.

The joint report to be submitted to the Authorities addresses the following topics:

- identification of the differences between the present state of the plant and the current safety rules and practices

- evaluation of the acceptability of these differences

- proposal for making appropriate improvements

- schedule for the implementation of the modifications.

Belgium has been the first nation in the world to have required these safety reviews on a periodic basis. Let us also note that the operating license does not fix a limit to the lifetime of the plant: the plant can operate as long as it stays in its safety domain, and periodic reviews are the means to ensure it for the next following ten years, in addition to the permanent surveillance exercised by the regulatory body.

As Doel 1 and 2 and Tihange 1 began operation in 1975, the first PSR took place for these plants in 1985. The modifications proposed were quite extensive. They have been realised for the most part in the period 1985-1990.

The PSRs of Doel 3 and Tihange 2, in operation since 1982, 1983, are now underway. In 1995 the second PSRs of Doel 1/2 and Tihange 1, will take place, as well as the first PSRs of Doel 4 and Tihange 3.

2. TYPES OF REVIEW DURING PLANT OPERATION.

Let us first reiterate some basic principles.

a. The operating organisation(licensee) is fully responsible for the safe operation of the plant

b. It must operate the plant in conformity with the requirements of its licence (in particular the Safety Analysis Report), under the supervision of the Authorities or of organisations delegated by the Authorities and acting on their behalf.

c. The licence contains a number of conditions like:

   - limiting conditions of operation,
   - periodic tests and inspections of safety related structures, systems and components,
   - procedures (technical and administrative) for plant modifications,
   - reporting of incidents, and resulting corrective actions,
   - feedback of operational experience from other plants.

   Thus nuclear power plants are subject to a process of continuous safety reviews throughout their operation. This process takes place at different levels in time:

   - ongoing routine surveillance during day to day operation,
   - tests and inspections of safety related components at periodic intervals; the periodicity depends on the component, varying from one week to 10 years.
   - proposals for modifications made by the licensee, reviewed and approved by the representatives of the Authorities. These modifications may be very minor ones or very important ones, like power increase, which requires a revision of the licence.
   - special safety reviews following accidents or incidents of safety significance which have occurred at the plant or elsewhere. These reviews are usually requested by the Authorities; examples are the application of lessons learned after the Browns Ferry fire or the Three Mile Island and Chernobyl accidents. Minor events brought to the attention of the regulatory body may also initiate a special review if the regulatory body so requests, feeling that their assessment should not be deferred to the next PSR.

   Due to the continuous surveillance of the safety of nuclear power plants, licensees in some countries have questioned the need of supplementary PSRs of the plants.

However, most licensing Authorities have thought that, complementary to this continuous surveillance, PSRs should be performed by the licensee for each nuclear power plant at suitable intervals during the plant lifetime, in order to get a broad integral view of the actual safety of the installation.

3. OBJECTIVES OF PERIODIC SAFETY REVIEWS.

The objectives of PSRs are to obtain a comprehensive assessment of the safety of the plant judged by current safety rules and practices, and using a stand back approach.

The objectives can be further detailed as follows:

a. to confirm that the plant is still at least as safe as originally intended, i.e. that no subtle degradation of the safety has taken place due, for example, to the accumulation of modifications made since the start-up.

   It might have been discovered during operation that some scenarios intended to be covered in the initial design, had in reality not been considered in some specific circumstances.

   The operational record of the plant and the operational feedback from similar plants are the main tools to perform this investigation. An evaluation of the modifications which have been made sheds also light on the safety or operation improvements really achieved versus the initial objectives of the modifications.

b. To establish the exact plant status and its operating experience with emphasis on those structures, systems and components susceptible to ageing and wear out. The aim is to identify and evaluate any factors which may limit the safe operation of the plant in the time interval till the next PSR.

c. To justify the current levels of safety of the plant by comparing them with current safety standards and practices and identify areas where improvements would be beneficial and risks reduced at justifiable expense.

The status of the plant is thus compared to the safety options adopted for the most recent units licensed, the design of which takes into account the evolution of the safety requirements and advances in technology to date. A further objective is to ensure a balanced approach to safety across the whole plant.

It should be pointed out that the surveillance of ageing and wear out processes is a permanent activity of the licensee. Before start up a qualification programme of safety related equipment has been set up, from which a Projected Qualified Life of each component has been deduced.

Preventive maintenance programmes take that information into account and replacements are scheduled accordingly. Sometimes, refinements in the evaluation of the Projected Qualified Life can extend this life (comparison between real and assumed environment, stresses, etc....).

Condition monitoring of the equipment and specific surveillance programmes (reactor vessel, steam generators,...) allow to follow most wear-out processes and apply the corrective measures in due time.

4. METHODOLOGY FOR THE PSRs.

Before starting the first PSR in Belgium, the experience already gained in other countries (France, Spain, systematic evaluation programme in the USA) was examined.

It was concluded that the PSR should be comprehensive, trying to review all the possible shortfalls of the plant with respect to current safety rules and practices.

In order to assess these shortfalls, the following steps were defined.

a. Identification.

- list systematically all topics where the status of the plant differs from what is requested for a new plant, using as a reference the contents of a Final Safety Analysis Report (e.g. as described in NRC Regulatory Guide 1.70 rev. 3 and the associated Standard Review Plans).

- list all rules and practices issued since the last PSR and examine their applicability.

- review the operating experience of the plant, with emphasis on incidents, modifications, periodic tests and inspections, technical specifications.

- review the operating experience of other plants of similar design, and the PSR experience of foreign countries.

- incorporate the results of a plant specific PSA, or of applicable generic PSAs.

- summarise the results obtained up to date from on going research programmes (severe accidents, human factors, ageing,...) and from the work on unresolved or generic safety issues, assess the degree of knowledge and decide what is mature enough.

This identification of subjects was made independently by the utility, its architect-engineer Tractebel, and by the regulatory body.

b. Selection.

The different topics identified by each organisation are collected and regrouped, and a preliminary list of subjects to be tackled during the PSR is established.
At this stage, the number of subjects to be re evaluated should not be considered as a limiting constraint. One should not scratch off subjects because they are a priori thought of minor importance or have already been considered more or less in depth in the past.

It is indeed important to be imaginative and to give a second look at subjects (incidents, modifications, improvements) tackled previously during operation of the plant with a broader perspective: have the objectives been reached, have unexpected difficulties been encountered, have the solutions adopted been effective while not introducing adverse side effects?

Moreover, consecutive to an incident, correcting the detected anomalies is by no means sufficient: root causes should be sought for and necessary action taken in order to prevent the recurrence of similar incidents taking place in the same or in more exacting circumstances.
Taking up these matters anew in the safety reassessment is a way to ensure that comprehensive solutions are adopted.

c. Study of each subject.

- the subject is clearly defined, explaining the initial status of the plant, and the aim of the study, i.e. the safety objective to be reached.

- the methodology is agreed upon (which kind of analysis to perform, what calculation to do, what initial conditions to assume in the safety analyses, etc....).

- the studies are performed by the architect-engineer or by the utility and submitted to the regulatory body for review and criticism.

- discussions are held between the various parties in charge in order to discuss the progress of each study, and reorient it if necessary.

- the conclusions of the studies clearly indicate the new level of safety aimed at and, as the case might be, the modifications needed to reach it.

- when the new level of safety is less than the one prescribed by current rules and practices, justification of the safety level retained is provided.

d. Integration of the studies.

- an integrated review of the results of all studies and proposed modifications is made in order to develop a global approach to the different problems and decide solutions mutually compatible. In this way it is possible to adopt modifications which are able to solve independently identified problems.

- the results of this first round of studies are the basis of the joint report submitted to the Authorities who endorse them with or without additional requirements.

e. Implementation.

- studies are refined for each subject, leading to the precise definition of the modifications to be made, their planning and implementation.

- the documentation (procedures, safety analysis report,...) are reviewed to make them reflect the modifications made and the results of the studies.

- the modifications are implemented at the plant, under the surveillance of the regulatory body.

5. THE FIRST PSRs OF DOEL 1/2 AND TIHANGE 1.

Doel 1 and 2 are Westinghouse twin plants of 400 MW(e) each; they are two-loop plants with an 8 ft core and a rather high average linear power (6,76 kW/ft or 221,8 W/cm). These two plants share a number of engineered safety features (ESF) systems:
4 high head safety injection pumps, 4 diesels, multiple interconnections between the 4 train electric buses, 4 component cooling pumps, etc. High head safety pumps inject water, up to a maximum pressure of 120 bar, in the cold loops and directly into the reactor vessel (upper plenum).

Tihange 1 is a three-loop Framatome reactor of 900 MW(e). It is similar to Beaver Valley in the USA and a forerunner of the Fessenheim plants in France, except that it is still fuelled with 15 x 15 type assemblies.
It has, like the 900 MW(e) plants of the French programme, two safety trains (2 x 100%) for electrical as well as mechanical equipment.
The high head safety injection pumps are the charging pumps, enabling them to inject water in the primary circuit at nominal pressure and even at higher pressure. Hence in case of a spurious safety injection signal, borated water will be injected in the core.

In the early 1970's when these plants were built less attention was given to the support systems for the safety systems; their seismic and post-accident environmental qualification was not a major concern; the protection against high energy line breaks was not considered for all systems, the physical separation between redundant systems was not so strict as it is in the more recent plants. Less attention was also given in these times to accidents from external origin, either natural or man made, like earthquakes, flooding, aircraft crashes, gas cloud explosion, toxic gases, large fires.

The safety reviews of Doel 1 and 2 and Tihange 1 were performed in parallel and the same subjects were considered for both plants, some of the subjects being more important or more difficult for one site than for the other one.

The list of subjects retained is given in Appendix 1.

This list was established in 1982-1983, as the plants began operation in 1974-1975, at a time when no probabilistic safety analysis of these plants was available. The comparison with at the time current rules and practices was based on the subjects considered during the safety analysis of the last four Belgian nuclear power plants Doel 3 and 4, Tihange 2 and 3 which were in their construction phase and where the current rules were followed.

This first PSR was made by applying deterministic safety criteria with the aim to bring the plants to the same level of safety as the plants under construction. No comprehensive probabilistic risk analysis was thought needed at this stage, but reliability studies of some of the systems were performed to help in the selection of the best improvements.

A subject not treated in this PSR was the simulators required for operator training because full scope simulators were being built for the new Belgian plants and their extension for the former plants was considered.

It is not possible to review in detail all the topics enumerated in Appendix 1, we will restrict ourselves to one example for Doel 1 and 2, and to another one for Tihange 1.

a. The addition of an emergency system at Doel 1 and 2.

This subject is developed in reference (1) and excerpts are given here.

From the analysis performed for the various subjects considered in the PSR it soon appeared that improving the existing safety systems would be very difficult for many of them due to the lack of space between redundant trains, due to the fact that earthquakes and external explosions had not been considered in the original design, due to the layout of the main steam and feedwater lines making high energy line breaks problems (in certain areas) about impossible to tackle.

It was therefore decided to add emergency systems and to house them in a new building, sharing with the existing systems just a few plant shutdown control components.

Designed to deal with major external explosion, earthquake, high energy line break outside containment, complete loss of on-site electrical supplies, major fire in the electrical building or affecting components not physically separated, the new emergency safety systems (one per unit) are installed in a new building and are able to stabilise the plant at hot shutdown conditions and to bring it to cold shutdown.

It should be noted that although multiple (4) electrical polarities have been used to achieve redundancy, most mechanical components are not redundant, consistent with a reliability objective of the emergency safety systems as a whole.

The safety functions of the emergency systems are:

. Safe shutdown of the reactor and residual heat removal
. Primary system integrity
. Radioactivity confinement and effluents control within authorised limits.

Detailed information on the systems and components can be found in (1).

The erection of the new building and the installation of new components took place in the period 1987-1990 when the plant was in operation as it did not interfere with it; the connection to existing systems and the integration tests were performed at the end of 1990 during a common shutdown of the two units.

It has been shown in this way that major improvements to cope with postulated initiating events not covered in the initial design can be made in a very efficient way and without interfering with plant operation.

b. The seismic revaluation of structures, systems and components at Tihange 1.

This subject is developed in reference (2) and excerpts are given here.

A safe shutdown earthquake (SSE) of 0,1 g had been considered in the design of Tihange 1. However during the experts review of Tihange 2, more conservative margins on the consequences of historical earthquakes led to the adoption of a SSE = 0,17 g.

For the reassessment of Tihange 1, it was felt more rational to adopt the new value of the SSE also for Tihange 1, which meant a 70% increase in the design earthquake intensity.

Better knowledge of the soil parameters and a site dependent design response spectrum were used in the reassessment of the buildings, using a three dimensional modelling. The results were satisfactory for the reactor building and for most of the other buildings, leading to some partial reinforcements of the upper structures. The electrical building however which is a framed structure without any shear walls could not resist the earthquake horizontal forces, using conventional analysis methods.
Reinforcement of the columns and addition of two longitudinal beams on the roof were analysed with a full three dimension non linear dynamic computer code specially developed and tested for framed structures, which showed these improvements were adequate.

In a second step representative piping systems were reassessed using the floor response spectra generated in the building revaluation. Use of non linear methods of analysis allowed to qualify the piping systems for seismic events practically in their original design conditions.

As a third step the qualification of active components was achieved using the Seismic Qualification Utility Group (SQUG) methodology developed in the USA in response to Unresolved Safety Issue A46.

Originally designed for a 0,1 g SSE, with less severe design criteria for structures and components than are presently required, Tihange 1 has been re evaluated for a SSE = 0,17 g.

The systematic use of classical methods could have led to extensive hardware modifications with the risk of safety impairment rather than safety improvement. But the use and development of innovative methods wherever applicable have allowed to qualify the plant for seismic events with only minor hardware modifications (like better anchorage of electrical cabinets) and thus reduced investments costs.

6. THE DOEL 3 AND TIHANGE 2 PSRs.

Doel 3 and Tihange 2 are three-loop PWR plants designed with an increased separation of functions between independent systems, and with three independent safety trains (mechanical and electrical).

Belgian safety authorities have required that the USNRC rules be followed and that these plants be protected against accidents of external origin like aircraft crash, explosions, large fires and toxic gases. These requirements have led to the addition of the so-called "second level protection" located in bunkered areas. It obeys the single failure criterion and consists in a three independent train additional protection system, pumps and tanks to inject borated water in the primary circuit, feed water for the steam generators, protection of the primary pumps seals, diesel groups, cooling systems and an ultimate heat sink.

Since these two plants have begun operation in 1982-1983, not many new safety rules have been published and the design of these plants compares quite well with the to-day evolutionary concepts for future reactors.
The reassessment thus draws heavily on the feedback of operating experience of the plants themselves, checking the consistency of the modifications performed during the last ten years, improving the test, maintenance and inspection activities, examining software, identifying precursor events from foreign plants experience, etc. It will also investigate combination of accidents not considered up to now, like a steam line break and a steam generator tube rupture at the same time, or the rupture of a few steam generator tubes. The list of the subjects retained for Tihange 2 is given in Appendix 2.

Some subjects are reviewed in more details hereafter:

6.1. The following examples relate to the first objective of § 3.

6.1.1. PSA

Since the Chernobyl accident, more emphasis has been placed on severe (beyond design) accidents, and investigations have been made on the ultimate strength of the containment, the hydrogen control, various ways to vent the containment, dilution scenarios which might lead to reactivity accidents, and possible accident management procedures.
A level 1+ PSA is also under way for the power as for the shutdown states. The results which are still preliminary indicate that these plants show a rather homogeneous safety level, with no scenario responsible for a large fraction of the core melt probability, and a positive impact of the "second level protection".

When compared to other plants, the importance of transients to the core melt probability is indeed smaller as the second level systems introduce effectively more diversity and/or redundancy.

The exact way in which PSA results will be incorporated in the decision process is still being discussed, but they will certainly be used in a relative manner to find out which systems play an important role in reducing the core melt probability, what gains could be achieved on their reliability, how allowed outage times could be more rationally defined in the Technical Specifications.

It is not intended to base decisions on the absolute value of the core melt probability and apply safety goals.
The Belgian safety philosophy remains the defence in depth (references 3 and 4): effectiveness of the barriers between the radioactive products and the environment, design against a set of postulated accidents. These criteria should not be jeopardised by too much credit given to absolute numbers resulting from PSA studies.

6.1.2. Post accident qualification of electrical connections.

Some of the electrical equipment located inside containment needs qualification to demonstrate that it is compatible with the harsh environment created by Design Basis Accidents.
Standards to this effect exist since the 70's, notably the IEEE-323.
Many tests have been performed during the construction phase of Doel-Tihange units. Whenever possible, existing test results were reviewed to ensure compatibility with the pressure-temperature profile calculated.

However, most test campaigns had been performed on single components; the important parameters had not always been measured, in particular, the loss of insulation resistance.
During an attempt to prove the overall functionality of essential equipment like motor-operated-valves and instrumentation sensors, considering of the electrical connections globally up to the primary containment, it was found that the minimum requirements for the insulation resistance may not be met.
It was thus decided to assemble prototype circuits and to submit them to a LOCA profile (after suitable ageing and irradiation) in a large vessel (the KALI loop in CADARACHE) in 1992. A variety of configurations were tested: terminal blocks, connectors, RAYCHEM sleeves, AUXITROL penetrations with KAPTON or PEEK.
The results confirmed the adequacy of some circuits, but also the excessive losses of insulation resistance when terminal blocks, unsealed, were used.
The plants modified progressively their equipment accordingly; short term procedures alleviated in the meantime the lack of full qualification.

6.1.3. Measurement of the free volume of the containment.

The reactor building free volume is an important parameter which intervenes in accident studies, for example to determine the resistance to pressure in the reactor building.
The value taken into account in accident studies was based on geometric calculations. It seemed useful to verify this value (and thus the validity of the accident studies) by a independent method.
An experimental method was developed which allowed to measure the free volume of the reactor building during the type A-test of the containment, with an accuracy of 2%. During the pressurising phase of the containment (prior to the integrated leak rate test), the free volume can be determined by measuring and correlating the injection flow of compressed air and the pressure increase in the containment. The tests performed at Doel 3 and Tihange 2 both confirm the geometrical determined values for the free volume of the reactor building.

6.2. The following example relates to the second objective of § 3.

Vessel embrittlement.
A new methodology based on the MCBEND Monte Carlo code was used for the determination of the RPV and surveillance capsule fluences instead of the usual 2D Sn multigroup calculations.
The accuracy increase of this new methodology is particularly due to an accurate 3D geometrical representation of the out-of-core regions and to the use of a point nuclear data library. The

statistical uncertainty on the small investigated volumes is now reduced taking advantage of the present computer performances and of the efficient variation reduction techniques implemented in MCBEND.
The neutron source distribution was deduced from measured flux maps and 2D calculations and was integrated along all the cycles.
This new methodology has led to a more accurate estimate of the present RPV fluence and to a more consistent set of capsules dosimeter results. This methodology was also used to extrapolate the RPV fluence for the next cycles.

6.3. The following examples relate to the third objective of § 3.

6.3.1. Clogging of recirculation sumps.

The risk of blockage of the recirculation sump screens by severely damaged insulation material has been recalculated. These screens had been designed using the models and hypotheses put forward in the Regulatory Guide 1.82 revision 0. Besides other requirements, this Regulatory Guide asked to determine head losses over the screen in case of 50% blockage of the screen, and to verify the respect of the NPSH required taking these head losses in account. The new revision of this RG required to determine real insulation debris volumes, following a model described in the related NUREG 897, which also put forward some formulas to calculate head losses for several types of insulation material. Since the RG and NUREG leave the user a lot of choice concerning the determination of calculation parameters and hypotheses, a lot of energy is put in discussions of the parameters and hypotheses to take in account. These discussions are still ongoing.

6.3.2. Post accident accessibility/Availability of low pressure safety injection pump during recirculation.

One has tried to improve the safety level of the plant in post accident conditions, by increasing the availability of the emergency core cooling (ECC) and containment spray (CS) pumps. Indeed, in the case of a LOCA, these pumps have to function for rather long time spans and multiple pump damages can occur. Two modifications have been carried out. In case of loss of all the ECC-pumps, an interconnection between the discharge of the CS-pump and the ECC-piping of the same train enables the CS-pump (having almost the same pump characteristic as the ECC-pump) to take over the core cooling function. If more time to intervene is available, provisions are present to flush and decontaminate the recirculation, CS- and ECC-piping, in order to reduce the radiation level and to allow interventions at the pumps. The borated water used to flush the piping is discharged inside the containment so that no spreading of contamination can occur.

7. THE NEXT PSRs.

In 1995, the second PSRs of Doel 1/2 and Tihange 1, and the first PSRs of Doel 4 and Tihange 3 will take place.

As all Belgian nuclear power plants have now comparable safety levels, these PSRs will probably be similar to those of Doel 3 and Tihange 2 underway. Probabilistic safety analyses have started for Doel 1/2 and Tihange 1. For Doel 4 and Tihange 3, the PSAs of Doel 3 and Tihange 2 will be used, taking into account the very small differences between these plants. The PSAs consider both power operation and shutdown modes.

8. CONCLUSIONS.

PSRs have been found a very efficient way to improve safety as summarised below.

The first safety reassessments performed for the Belgian plants have resulted in a very extensive re analysis of the safety of the plants, and the adoption of mutually consistent modifications able to provide integrated solutions to many different problems. Such a global approach permits a much more efficient safety reassessment and improvement capability than a case by case analysis and curing.

It has been shown that improvements could be made even for topics very deeply linked to the design and the lay-out of the plant itself, like protection against external hazards, increase in the intensity of the design basis earthquake or modification of the reactor protection system.

When there are limitations to the upgrading of existing systems, a solution is to add dedicated systems which will ensure the safety functions sought, as it has been shown with the ultimate shutdown system at the Doel plant. It is thus possible to extend the design bases of the plant to cover events not considered at the onset.

The end result is the upgrade of the safety of the older plants towards a safety level similar to the one of the most recent plants.

Once a robust safety case has been obtained through a strict adherence to the deterministic safety principles, a probabilistic safety analysis is appropriate due to its systematic approach and the different light it can shed on the installation, leading to a more refined analysis and further enhancement of safety.

Periodic safety reassessments are viewed as the best way to guarantee the safety level of the existing plants and to incorporate in an ordered and comprehensive manner the evolution of the safety principles on which the design of the most recent plants is based. It is also a good way to ensure continuity of knowledge about the safety of the plant when turnover of the personnel occurs.
It should be noted these conclusions are in line with those reached in other European countries, as shown in reference (5).

REFERENCES

1. B. Danhier
   The 10 year safety reassessment of Doel 1/2. Addition of an emergency safety system.
   BNS annual meeting, May 1990, Brussels.

2. M. Damsin
   The 10 year safety reassessment of Tihange 1, and the seismic re evaluation of the structures, systems and components.
   BNS annual meeting, May 1990, Brussels.

3. P. Govaerts
   The evolution of safety measures in Belgian nuclear power plants.
   IAEA Conference "Nuclear Power Performance and Safety".
   Vienna, October 1987 (paper IAEA-CN-48/149).

4. P. Govaerts, H. Dresse, M. Roch
   Upgrading requirements.
   Proceedings of the ENS Topform Conference
   Prague, October 1992.

5. Periodic Safety Reviews of Nuclear Power Plants in EC Member States, Finland, Sweden and Switzerland. A review of current practices.
   D. Goodison & al. (EUR report in preparation).

APPENDIX 1

SUBJECTS RETAINED FOR THE FIRST PSRs OF DOEL 1/2 AND TIHANGE 1.

The following subjects have been considered:

1. Protection against accidents of external origin and industrial risks.
2. Earthquakes.
3. High energy line breaks.
4. Fire protection.
5. Flooding from external or internal origin.
6. Large winds, climatic effects.
7. Differential settling of buildings.
8. Systems to stop the reactor, cool the core, and remove the residual heat:
   - reactor protection system
   - safety systems: auxiliary feed water to the steam generators, shutdown cooling, safety injection, spray (ventilation of the reactor containment), control room and auxiliary shutdown panel
   - steam discharge to the atmosphere
   - ultimate heat sink
   - safety related pressure air
   - electric supplies
   - resistance and integrity of the circuits
   - instrumentation for the safety related systems
   - leak detection for the primary circuit
   - detection of inadequate core cooling
   - qualification (seismic and environmental) of electrical and mechanical systems
9. Primary circuit integrity:
   - over pressure protection in hot and cold conditions
   - protection against pressurised thermal shock
   - reactor vessel venting
   - seals integrity of the primary pumps
   - leak detection
   - corrosion due to boric acid
   - list of transients which have occurred
10. Nuclear auxiliary buildings: accessibility and equipment qualification after a severe accident
11. Inspection of structures and equipment (electrical, mechanical)

12. Test programme
13. Technical specifications
14. Conduct of operation
15. Quality organisation
16. Spent fuel handling and storage
17. Ventilation systems, gaseous waste system
18. Isolation and leak tightness of the primary and secondary containments
19. Hydrogen control in the primary containment
20. Operation experience
21. Accident analysis
22. Radiation monitoring and ALARA policy
23. Post accident sampling in the reactor building, post accident radiation shielding after core degradation
24. Updated documentation, including revision of the FSAR.

APPENDIX 2

SUBJECTS RETAINED FOR THE FIRST PSR OF TIHANGE 2 (DOEL 3)

This list is not exhaustive, as subjects of very narrow and specific scope have been omitted, for the sake of conciseness.

Environment

1. Reassessment of the industrial environment
2. Protection against electrical shocks and extreme climatic conditions.

Design criteria

3. Impact of modifications on original HELB studies.
4. Qualification of mechanical components, in particular those of limited lifetime.
5. Anchorages of components.
6. Dimensioning of no flow lines for centrifugal pumps.
7. Post accident operability of pneumatic valves.
8. Relief and safety valves of the pressuriser, and their discharge line.
9. Thermal environment of electrical components.
10. Post accident qualification of electrical connections.
11. Review of new U.S. rules and practices.

Core design

12. Reloads safety analysis.

Primary circuit

12. Follow-up of transients during operation.
13. Drift in safety valves set points.
14. Vessel embrittlement.
15. Thermal ageing of austenitic steels.

Engineered safety features

16. Clogging of recirculation sumps.
17. Chemistry of spray water.
18. Measurement of the free volume of the containment.
19. De pressurisation of safety injection accumulators.
20. Availability of low pressure safety injection pumps during recirculation
21. Manual actuation of the containment spray system.

Instrumentation and control.

22. Core thermocouples
23. Response time verification for sensors.
24. Diesels availability during intricate scenarios.

Electrical systems.

25. Electric motors in degraded voltage conditions.
26. Loading of the Diesel generators.
27. Procedures dealing with the loss of low voltage busses.

Systems.

28. Leak tightness of seals between fuel storage pools.
29. Differential pressures in ventilation systems.
30. Fire protection.

Radio protection.

31. ALARA policy.
32. Assessment of radio protection techniques.
33. Post accident accessibility.

Operation.

34. Review of the programme for operator training and licensing.

Tests and inspections.

35. Leak tightness of recirculation lines.
36. Snubbers tests.
37. Review of periodic tests (pumps, valves, check valves).
38. Pressure vessel, safe ends: under clad defects.
39. Corrosion and wear-out of various components.
40. Inconel in the primary circuit.
41. Evolution of the ASME XI code, including new appendices.

Accidents studies and PSA.

42. Limit value of the primary/secondary leak.
43. Accidents not considered in the original design.
44. Severe accidents.
45. Probabilistic safety analysis.

Technical Specifications.

46. Evaluation of the Technical Specifications.

Quality Assurance.

47. Quality Assurance programme: experience gained.
48. Software quality assurance.

Feedback of operating experience.

49. Review of the incidents and root cause analysis.
50. Assessment of modifications impacting safety.
51. Review of the emergency (second level) protection systems.
52. Operator aids in shutdown modes and after an accident.
53. Primary break in modes 3 and 4.
54. Thermal stratification in the surge line and in feed water lines.
55. Assessment of voluntary inspections.