OPERATING EXPERIENCE WITH THE LATINA MAGNOX REACTOR

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Abstract

The paper describes the plant and gives the relevant data concerning its operational life. Construction of the Station started in 1958, first criticality occurred in Dec. 1962, first connection to the grid in May 1963, and commercial operation started from Jan. 1964.

The nominal electrical output of 200 MW was reduced in 1969 to 160 MW to cope with the oxidation of the primary circuit mild steel structures caused by hot CO2, similar to all the steel-vesselled Magnox stations.

Periodic inspections of components in the primary circuit were thus scheduled to check for the integrity and present damages.

Repairs were carried out to assure safe operation. In Nov. 1986 the Station was shut-down for planned overhaul and after the national referendum (Nov. 87) called in by the post-Cernobyl mass criticism towards nuclear generation, the Italian Government decreed its final shutdown (Dec. 1987) and the owner was ordered to start actions to bring the plant in a passive protective condition (equivalent to IAEA Phase 2 of decommissioning). The governmental decision is political and not based on safety grounds.

The operating licence, renewed in 1985 by the Italian Ministry of Industry, was due to expire in 1992, with provisions of possible extension.

Renewal in 1985 was obtained after a full safety review made with the same ground-rules of CEGB Magnox Stations. It was, in its completeness, an anticipation of CEGB’s Long Term Safety Review for 20 years operation. Only very limited backfitting was considered necessary; in particular a new emergency feed water system was built and modifications of safety circuits were undertaken.

Also a first-level Probabilistic Safety Assessment was carried out, and areas where simple plant modifications could give significant improvement in overall safety were found.

The cumulative electrical output has been 26 TWh, scoring an average availability factor of 73%.

A description of the inspections to the primary circuit internals and repairs is detailed.

Other unusual operating events and radiation protection data are also outlined.

Conclusive experience of the 75 years of life shows that the operation of the Latina Magnox has been regular and in accordance with high safety standards.

The reactor is presently undergoing the Phase 1 decommissioning (de-fuelling) which is to be completed in 3 years.

1. DESCRIPTION OF THE PLANT

The Latina Station, on the coast of the Tyrrhenian Sea, 70 km south of Rome, is equipped with one Magnox reactor. The plant was built as a joint venture between the British Nuclear Power Plant Company Ltd (BNPPC) and Italian Agip Nucléare (AN). The time schedule was as follows:

- Aug. 1958 : date of contract;
- Oct. 1958 : beginning of construction;
- Feb. 1961 : vessel pressure test on site;
- Dec. 1962 : first criticality;
- May 1963 : first connection to the grid;
- Jan. 1964 : start of commercial operation.
The general outline of this Magnox plant is derived from the Calder Hall prototype (fig. 1).

The reactor building is separate from the conventional part of the plant and contains the vessel with the core, the primary coolant circuit, with 6 heat exchangers and 6 gas circulators, and the concrete biological shield (fig. 2).

The turbine hall contains 5 sets with the steam cycle and the plant electrical services switchgear.

The control building is in the middle, with the main control room and the essential electrical supply system.

The reactor building is directly connected to the irradiated fuel route and to the cooling pond, with its water treatment plant and liquid effluent system.

The core consists of a 24-sided vertical prism made of graphite blocks in layers, bored and assembled to form vertical cylindrical channels for fuel, control rods and steel neutron absorbers. The total weight of the core with fuel is about 2500 t. Each fuel channel holds a uranium metal rod, each in a finned Magnox can (polyzonal type). The radial neutron flux distribution is controlled with vertical steel absorbers suitably inserted in interstitial channels.

The primary circuit is made by a spherical carbon steel vessel, 20 m dia. and 6 steel ducts, 2 m dia.

The nuclear reaction is controlled by insertion from the top of 88 boron steel rods in the core. In addition, 12 safety rods are provided. The rods are suspended on chains directly into the stand-pipes which penetrate the pressure vessel. 20 rods are in automatic control.

The coolant is CO2 in closed circuit; it goes through the core from bottom to top. The heat exchangers, one per each duct, are placed high enough to allow for natural circulation.

The coolant is forced into the sphere by 6 axial circulators, each driven by an AC variable speed motor. Auxiliary motors are operated by normal and the emergency electrical system.

The pressure vessel is surrounded with a cylindrical concrete biological shielding, 3.20 m thick average. Another concrete wall 1.65 m thick protects the gas duct outlets.

Fuel handling is on-load by means of a charge-discharge machine, carried on a gantry on the pile-cap. The spent fuel withdrawn from the reactor, is directly discharged into the cooling pond before its transport to the re-processing plant.
A dual thermal cycle (HP and LP) operates three sets of 70 MW turbogenerators and two auxiliary variable speed turbogenerators for powering the coolant circulators.

The condensing water, in open circuit, is taken from the sea at a pier 700 m long for deeper waters.

The plant main technical data are listed in tab.1.

2 - BEHAVIOUR OF THE PLANT

From the beginning (1963) to the last shut-down (1986), the overall gross generation of the plant has been 26 TWh (electrical) with an average availability factor of 73%. It must be taken into account that the plant belongs to the first generation of nuclear power stations, and that at the time it was the first reactor with an output of 200 MWs in Europe.
In 1963 Latin* was the best European nuclear station for the annual availability factor (96%).

The yearly generation is shown in fig.3.

The longer outage periods are described below.

- In 1969 a 207 days outage was necessary to inspect the inner structures of the reactor and of the six steam generators to check the effects of the oxidation on mild steel components, caused by CO2 at high temperature. Details of the consequences produced by this common feature for the steel - vesseled Magnox plants on Latina Station are in par. 3 and 4.

Following these inspections, it was decided to lower the outlet CO2 temperatures from 390°C to 260°C with an output loss from 210 to 160 MWe (-24%). During the following start - up of the plant a number of failures in fuel elements occurred (see par. 6), particularly at mid - channel position. The shut - down was prolonged and a period of low power in order to detect and discharge the damaged fuel followed.

- In 1973 the plant was shut - down for 107 days to remove the last two stages of blades on the LP section of the main turbines, due to suspected cracks on the key - way at the shaft. The output was subsequently reduced to 150 MWe. Later a new set of LP rotors was replaced and the output was set back to 160 MWe. During this shut-down the repair of the thermocouple cables securing system on the charge - pans was completed in the reactor (see par. 4).
- In 1979 the planned overhaul was extended for 139 days for inspections on internal and external components of the primary circuit (mainly welds on the ducts bellows).

After 1989, the Regulatory Authority compelled the owner to plan a reactor internal inspection to the mild steel structures operating at high temperature every year; only after 1979, evidence on the slow development of the oxidation and on the effectiveness of modelling was such to allow an interval of two years for the inspections.

- In 1986 the Station was shut down for the planned overhaul. The scheduled activities and controls ended in Dec. 1986; the inspections to the damaged links connecting the internal restraint structure to the core (see par. 4) continued until March 1987 and the station was kept shut down pending a national referendum on some laws about nuclear power installations, called in by the emotion of the Chernobyl accident.

The referendum took place in November 1987, with disastrous effects on nuclear generation in Italy, and in Dec. 1987 the Italian Government decided to close the plant and the owner was ordered to start actions to bring the plant in a passive protective condition although the operating license, renewed in 1985 by the Italian Ministry of Industry, was valid until 1992 (thirty years from first criticality), with provisions of possible extension subject to a full plant safety review.

The plant has so started discharging the irradiated fuel, and arrangements for shipment to the reprocessing plant of the full core are already made. Presently (Sept. 86) about 10% of the core has been discharged, and completion is expected in three years.

3. OXIDATION OF MILD STEEL STRUCTURES IN THE PRIMARY CIRCUIT

The oxidation phenomena caused by hot CO₂ in Magnox reactors are known since the late sixties. At Latina, during the first inspection in 1969, some damaged parts were found in the heat exchangers (hangers, bolts, gas deflectors) and on the reactor top charge pans (T/C cable holders).

The actions taken to continue a safe operation were as follows.

i) Periodical inspections to the primary circuit mild steel structures, to check any failures or defects insuring during previous operation.

For details see par. 4.

ii) Reduction of the maximum gas outlet temperature of 30°C in order to reduce the oxide growth rate. This figure was common to the Magnox steel-vesselled reactors, and came from a compromise between the necessity of stopping the oxide growth and the operational difficulties caused by a too dramatic reduction in temperature to the conventional machinery.

iii) Closer control on the coolant chemistry. It was soon established that the oxide growth rate on mild steel surfaces is mainly due to hot CO₂, but it is also influenced by residual gases present in the coolant, in particular by H₂O vapour and H₂ in a negative way, and CO in a slightly positive way.

The actions to obtain significant reductions in water content were:

- more stringent specifications for fresh CO₂ external supply;
- improvement of the drying system of fresh CO₂ produced at the plant;
- control instrumentation more sensitive to low values;
- improvement in purging procedures for the fuel machine before coupling to the reactor.

The actions to keep H₂ content the lowest possible were:

- more stringent limits in fresh CO₂ supply;
- more careful operational checks and maintenance to CO₂ sealing on gas circulators (oil-operated).
### TABLE 2. COMPONENTS INSPECTED AND METHODS OF INSPECTION

<table>
<thead>
<tr>
<th>COMPONENT</th>
<th>METHOD</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas outlet ducts, thermal insulation</td>
<td>TV cc</td>
</tr>
<tr>
<td>3&quot; and 1&quot; bolts, core restraint structure upper layers</td>
<td>Ultrasonics</td>
</tr>
<tr>
<td>Vessel thermal insulation (int. el.) and thermouples</td>
<td>TV cc, photographs</td>
</tr>
<tr>
<td>Links of the core restraint structure</td>
<td>TV cc</td>
</tr>
<tr>
<td>Vessel welds and stand-pipe forgings (external)</td>
<td>Photographs</td>
</tr>
<tr>
<td>Charge pans</td>
<td>Magnetic particles,</td>
</tr>
<tr>
<td>Duct bellows welds</td>
<td>Ultrasonics</td>
</tr>
<tr>
<td>Boiler manholes welds</td>
<td>Ultrasonics</td>
</tr>
</tbody>
</table>

### 4. PERIODICAL INSPECTIONS AND REPAIRS

Starting from 1969, periodical inspections of the primary circuit components were carried out, as indicated before, particularly to check the progress of the oxidation on mild steel structures and the possible effects of ageing on materials (specially welds).

Methods of inspections ranged from qualitative (using TV cameras in closed circuit and photographs, magnetic particles and dye-penetrants) to quantitative (ultrasonics). For a list of components inspected and the methods of inspection see tab. 2.

The frequency of primary circuit inspections was every year until 1979, and then every two years. As it requested depressurization of the circuit and cooling of the reactor for every inspection, it had a significant effect on plant availability.

Also samples to check the oxidation rate were periodically extracted from or measured in the heat exchangers and tests for accelerated oxidation rate were carried out in rigs at CISE Laboratories in Milan.

Every two years since 1973, a detailed assessment of the safe condition of reactor internals operating at high temperature was produced for the Regulatory Authority. The oxide growth rate was assessed and predicted according to a general model provided and updated by CEGB, valid for all the Magnox stations. The model was set according to a series of theoretical studies, experiments and to a great amount of sampling from the primary circuits.

The result was to produce a set of failure probabilities of components in the future operation, to be compared with the real failure frequencies resulting from the inspections.

The results of the ultrasonic inspections to the restraint structure bolts of the higher layers (highest operating temperatures) up to 1986 showed a small percentage (≤ 1%) of failures, well in agreement with the failure predictions obtained with the periodical safety assessment. Other internal components were in good conditions. Welds on bellows, on ducts, on manholes of the primary circuit showed few original defects, well below the acceptable dimensions and not growing during operation.

As quoted above, during the reactor internal inspections in 1969, some clamps and bolts fastening the thermocouple cables for gas outlet temperature measurements were found broken or out of position due to oxidation. A mis-positioning could disturb the fuel operations and debris in the core channels reduce cooling. So it was decided to sleeve the channels on the charge pan top plate with cylindrical steel rings. The thermocouple junctions were also clamped in position. This repair was completed in 1973.

Repairs were also carried out on cracked welds on the blowers deflectors. It was a much more easy task, the area being directly accessible.

During the last planned overhaul for inspections also failures on the links connecting the graphite core and the restraint structure beams were found. This requested a more detailed investigation with a photo camera that reached the bottom layers of the structure (see fig. 4).

The expected cause of the failure can be ascribed to the seizure of the spherical joints at both ends, due to oxidation of the surfaces.
The number of failures and defects found on these components, although it can be demonstrated not being a safety problem, compelled the operator to envisage the restoration of the function. The architect-engineer, NNC, provided a design for replacement of these components. This activity did not seem, at that moment, to require complicated tools, to be very expensive and time-costly. Repairs could be easily carried out by station Personnel in a few months.

5. BACK - FITTING

Only few improvements on the safety-related systems had to be carried out during the first operating years.

After the overall safety review, completed in 1977, the Regulatory Authority requested in 1979 a further reassessment of the plant using any new safety criteria and experiences applicable to gas reactor (post THI plant review).

The main requests concerned:

- revision of the plant response to original design basis accidents, taking into account all the plant modifications completed in the previous operation, ageing, operational experience, up-dating of models and safety ground-rules;
- definition of safety-related systems and components that emerged from the previous point;
- up-grading of the technical documentation;
- implementation of QA for safety-related plant modifications;
- up-dating of operating rules;
- requalification of radiological impact under normal and accident conditions;
- up-dating of emergency planning.

The Regulatory Authority asked also an analysis or a reassessment of events not considered explicitly during plant construction, the most relevant of which were earthquake, tornado, pipewhip, LOOSEP, fire, man-made events.

Most of them were already well covered by the original design and construction. For earthquake, see par. 9.

The review was carried out with the collaboration of CEGB, using the same criteria and groundrules, when applicable.

The plant modifications that were considered necessary according with the results of the reviews are listed in tab.3.

6. FUEL AND COOLING POND BEHAVIOUR

The irradiated fuel elements are kept under water in a pond for radioactive decay. Before the periodical shippings of the fuel to the reprocessing plant (Windscale), each element is desplitted to reduce volume.

The pond water is treated in closed circuit with ion-exchange resins. The regenerants of the resins are sent to an inorganic exchange resin for Cs 134+137 concentration. The water is also sand-filtered and re-
TABLE 3. PLANT MODIFICATIONS

<table>
<thead>
<tr>
<th>Before 1979:</th>
</tr>
</thead>
<tbody>
<tr>
<td>- maximum fuel temperature protection modification</td>
</tr>
<tr>
<td>- seismic trip system installation</td>
</tr>
<tr>
<td>- boron carbide powder ultimate protection installation</td>
</tr>
<tr>
<td>After 1979:</td>
</tr>
<tr>
<td>- emergency CO₂ supply system improvement</td>
</tr>
<tr>
<td>- emergency Diesel generators improvement</td>
</tr>
<tr>
<td>- blowers house segregation</td>
</tr>
<tr>
<td>- access control implementation for security</td>
</tr>
<tr>
<td>- installation of a new separate emergency feed-water system</td>
</tr>
<tr>
<td>- modification of trip logics for dp/dt</td>
</tr>
<tr>
<td>- liquid effluent system improvement</td>
</tr>
<tr>
<td>- feed low pressure protection installation</td>
</tr>
</tbody>
</table>

frigerated. The behaviour of fuel handling equipment must be considered satisfactory.

The channel irradiation discharge limit was originally 3600 $\text{MWD} \text{t}^{-1}$ with a dwell-time limit of 6 years.

In 1979 an improved type of fuel for a higher burn-up was loaded in the reactor, allowing to increase progressively the burn-up to 5000 $\text{MWD} \text{t}^{-1}$ and the dwell-time to 11 years, with no operational problems.

The number of damaged fuel elements up to 1968 was in average 15 per year. In 1969 and 1970 many elements were discharged from the reactor after a long shut-down. So in 1970 it was decided to put a fuel element of the high-temperature type in fourth position in the channel, instead of the low-temperature type. This significantly reduced the number of burst elements to nil in practice (see fig. 5).

The spent fuel discharged in the pond experienced severe corrosions in four periods, with significant increase of contamination levels in the water. It occurred when de-splitted elements had to be kept for long time in the pond. With a more and more stringent control of the chemical and physical conditions of the pond water, low levels of contamination are now obtained.

The quantity of fuel discharged from the reactor and sent to the reprocessing plant from the beginning up to the end of 1987 is 1,170 tons.

7. RADIATION PROTECTION

A - Active effluents

The gaseous effluents are produced by activation of the open-circuit cooling air of the biological shield and of the vessel, by coolant controlled releases to the ambient due to depressurization of the fuel discharge machine or of the primary circuit and by particulates that may escape from the filters on the discharge lines.

Annual limits are set by the Regulatory Authority, for halogens, noble gases and beta-gamma emitting particulates.
### TABLE 4. ANNUAL ACTIVITY RELEASES AND PERCENTAGE OF ANNUAL ALLOWED LIMITS

<table>
<thead>
<tr>
<th>GASEOUS</th>
<th>LIQUID</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Activity Release</strong></td>
<td><strong>% of annual allowed limits</strong></td>
</tr>
<tr>
<td><strong>Activity Release</strong></td>
<td><strong>% of annual allowed limits</strong></td>
</tr>
<tr>
<td><strong>Year</strong></td>
<td><strong>A &amp; B</strong></td>
</tr>
<tr>
<td>75</td>
<td>10</td>
</tr>
<tr>
<td>76</td>
<td>16,3-10^17</td>
</tr>
<tr>
<td>77</td>
<td>16,3-10^17</td>
</tr>
<tr>
<td>78</td>
<td>16,3-10^17</td>
</tr>
<tr>
<td>79</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>80</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>81</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>82</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>83</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>84</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>85</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>86</td>
<td>12,1-10^17</td>
</tr>
<tr>
<td>87</td>
<td>12,1-10^17</td>
</tr>
</tbody>
</table>

**Limits:**

- (A) 10^14 Bq
- (B) 10^12 Bq
- (C) 10^10 Bq

In Table 4 are shown the annual activity releases for the most relevant radionuclides (A41) and the percentage of the annual allowed limits covered for noble gases, halogens, and particulates starting from 1975.

The liquid effluents are produced 95% from the treatment plant of the spent fuel pond water, and 5% from the decontamination facilities on site and the active laundry. They are released to the sea water at the discharged culverts. The values for the annual discharge of H3, all the other radionuclides, and the percentage of the annual limits covered by the liquid discharge are shown in Table 4.

The limits are apparently very strict compared with other similar plants on an open sea. It must be considered that the Latina front basin is very shallow.

The solid wastes take origin from decontamination, coveralls, contaminated solid materials, fuel element splitters, mud from the pond water treatment plant, depleted inorganic ion exchange resins. These wastes are stacked on site, the low activity in carbon steel 220 litres drums, the splitters and other high activity solids in underground bunkers.

The annual production rate is of about 50-100 drums for low activity, 10-20 m³ for splitters and 1-2 m³ for mud.

**B - Environment surveillance**

A local network for environment samples is operating around the Station. The samples are listed in Table 5. The results of the measurements are published every year, and the contamination of the samples due to the Station effluents is not detectable since the beginning.

**C - Dose to critical groups of population**

The assessed dose (whole body) from the gaseous discharges to the critical group of the population (local farmers) can be seen in Table 6 from 1974 to 1986.

The assessed dose (whole body) from the liquid discharges to the critical group (local fishermen) is in Table 6 from 1974 to 1987.

The results show that the environmental impact of Latina station has always been negligible.

It must be also considered that the assumptions for the modelling are considerably pessimistic.
### TABLE 5. ENVIRONMENTAL SAMPLES

<table>
<thead>
<tr>
<th>Type of Sample</th>
<th>No of Samples</th>
<th>Analysis Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 - Air</td>
<td>4</td>
<td>Thrice a Week</td>
</tr>
<tr>
<td>2 - Pull-out</td>
<td>2</td>
<td>Continuous</td>
</tr>
<tr>
<td>3 - Green</td>
<td>3</td>
<td>Semester</td>
</tr>
<tr>
<td>4 - Goat or cow milk</td>
<td>1</td>
<td>Semester</td>
</tr>
<tr>
<td>5 - Eggs</td>
<td>1</td>
<td>Semester</td>
</tr>
<tr>
<td>6 - Vegetables</td>
<td>1</td>
<td>Semester</td>
</tr>
<tr>
<td>7 - Fruit</td>
<td>1</td>
<td>Annual</td>
</tr>
<tr>
<td>8 - Sand/sediments</td>
<td>1</td>
<td>Monthly</td>
</tr>
<tr>
<td>9 - Sediments</td>
<td>1</td>
<td>Quarterly</td>
</tr>
<tr>
<td>10 - Periphyton</td>
<td>1</td>
<td>Annual</td>
</tr>
<tr>
<td>11 - Fish</td>
<td>1</td>
<td>Quarterly</td>
</tr>
<tr>
<td>12 - Rainwater</td>
<td>1</td>
<td>Semester</td>
</tr>
<tr>
<td>13 - Salt Water</td>
<td>2</td>
<td>Continuous</td>
</tr>
</tbody>
</table>

### TABLE 6. ASSESSED DOSES TO CRITICAL GROUPS

<table>
<thead>
<tr>
<th>Year</th>
<th>Liquid, mSv (Professional)</th>
<th>Gaseous, mSv (Local farmers)</th>
</tr>
</thead>
<tbody>
<tr>
<td>74</td>
<td>2.3 \times 10^{-2}</td>
<td>4 \times 10^{-3}</td>
</tr>
<tr>
<td>75</td>
<td>1.7 \times 10^{-2}</td>
<td>2 \times 10^{-3}</td>
</tr>
<tr>
<td>76</td>
<td>1.7 \times 10^{-2}</td>
<td>2 \times 10^{-3}</td>
</tr>
<tr>
<td>77</td>
<td>1.3 \times 10^{-2}</td>
<td>2 \times 10^{-3}</td>
</tr>
<tr>
<td>78</td>
<td>8 \times 10^{-3}</td>
<td>7 \times 10^{-3}</td>
</tr>
<tr>
<td>79</td>
<td>1.6 \times 10^{-2}</td>
<td>6 \times 10^{-3}</td>
</tr>
<tr>
<td>80</td>
<td>4 \times 10^{-3}</td>
<td>6 \times 10^{-3}</td>
</tr>
<tr>
<td>81</td>
<td>6 \times 10^{-3}</td>
<td>6 \times 10^{-3}</td>
</tr>
<tr>
<td>82</td>
<td>1.1 \times 10^{-2}</td>
<td>6 \times 10^{-3}</td>
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<td>83</td>
<td>3 \times 10^{-3}</td>
<td>9 \times 10^{-3}</td>
</tr>
<tr>
<td>84</td>
<td>2.8 \times 10^{-3}</td>
<td>4.2 \times 10^{-3}</td>
</tr>
<tr>
<td>85</td>
<td>2 \times 10^{-3}</td>
<td>5.9 \times 10^{-3}</td>
</tr>
<tr>
<td>86</td>
<td>6.6 \times 10^{-3}</td>
<td>1.3 \times 10^{-3}</td>
</tr>
<tr>
<td>87</td>
<td>1.3 \times 10^{-3}</td>
<td>/</td>
</tr>
</tbody>
</table>

D - Doses to the workers

Starting from 1963 up to 1987, in tab. 7 are indicated the collective dose, the average and the maximum annual individual dose both for the operator's staff and for workers of outside firms. It can be noted that a considerable effort was taken to keep down the collective dose, although very low, improving from experience the quality of the work in the controlled areas with remarkable results.

## B - UNUSUAL OPERATING EVENTS

Unusual operating events that affected the plant during its life, besides the reactor internal structures oxidation and damages, and the main turbines LP rotor replacement, are shortly listed below. These unusual events did not affect significantly the availability of the plant, and generally only simple remedial actions were necessary.

- LOCSP for 40 min in 1978. It caused a reactor scram, and the emergency Diesel concerned regularly started automatically to supply essential loads. Remedial action: new set of the national grid connected to the plant and better coordination of line protections.

- Loss of minimum stock of fresh CO2 on site. The fresh CO2 is stocked on site in liquid phase in drums. The failure of a burst disc caused by frost produced a loss of fresh CO2 below the minimum content allowed by the operating rules.

  Remedial action: periodic check of instrumentation and valves concerned.

- Corrosion from water dipping of a CO2 reactor filling pipe. Remedial action: full inspection of similar pipes, repair of the failed water drain that caused the dipping.
### Table 7. Collective Dose to Workers

<table>
<thead>
<tr>
<th>Year</th>
<th>N° of persons</th>
<th>Collective dose (man Rem)</th>
<th>Average individual (dose Rem)</th>
<th>Maximum individual (dose Rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Staff All</td>
<td>External</td>
<td>Staff All</td>
<td>External</td>
</tr>
<tr>
<td>1986</td>
<td>33</td>
<td>33</td>
<td>0.33 0.33 0.33</td>
<td>0.33 0.33 0.33</td>
</tr>
<tr>
<td>1987</td>
<td>33</td>
<td>33</td>
<td>0.33 0.33 0.33</td>
<td>0.33 0.33 0.33</td>
</tr>
<tr>
<td>1988</td>
<td>33</td>
<td>33</td>
<td>0.33 0.33 0.33</td>
<td>0.33 0.33 0.33</td>
</tr>
<tr>
<td>1989</td>
<td>33</td>
<td>33</td>
<td>0.33 0.33 0.33</td>
<td>0.33 0.33 0.33</td>
</tr>
</tbody>
</table>

- Failure of bolts connecting some stand-pipes assemblies to the fuel element guide tube, caused by fatigue failure. Remedial action: replacement of the bolts.

### 9. Seismic Analysis

Following a request of the Regulatory Authority in 1979, a seismic analysis was carried out for the Latina site.

Originally the most relevant parts of the plant were designed against earthquake taking into account a static acceleration of 1/8 g.

A study made on historical documents demonstrated that the maximum foreseeable earthquake on the site was not more than VI-VII MCS (Mercei, Giampari, Sielberg) with a return time of more than 1000 years.

This corresponds to a peak acceleration at ground level of 0.07 g.

However, in order to increase the knowledge of the site environment, a network of sensors was installed around the area in order to record the local microseismicity. In a couple of years, no significant local activity was recorded, confirming the general knowledge of the area being a "quiet" one from the seismic point of view.
To take into account all sort of possibilities, the final Authority's position was that all the safety related structures and systems had to withstand an earthquake of VII-VIII MCS with a corresponding peak acceleration of 0.1 g. Calculations were then carried out with the dynamic spectrum as per RG 1.03 scaled to 0.1 g.

Agreement was found with the Authority on the possibility that local ductility effects could be called in for an acceleration limit down to 0.07 g. Below this value, the operator had to provide modifications to the structures.

According to the calculations made with these inputs the major modifications concerned these items:
- main and auxiliary turbines hall roof
- liquid radwaste building
- boilers supports.

It was mainly civil work, and was left to the preliminary design stage.

10. RISK ANALYSIS

After the periodical safety review of the plant, the Regulatory Authority requested to the owner to undertake a probabilistic safety study (P.S.S.), with the object of assessing the overall safety of the plant, and of detecting the accident conditions that would damage the fuel, with a measure of the associated frequency.

The study was commissioned by the National Nuclear Corporation Ltd (NNC), and it took about one year to be completed. The study identified and quantified the frequency of the fault sequences which could lead to core damage, in order to determine the dominant sequences and to identify possible plant modifications, in addition to those already defined, to further improve plant safety.

These conclusions emerged:

(i) The frequency of sequences outside the design basis with the potential for core damage due to failure of the DHR systems is predicted to be $1.4 \times 10^{-2}$ for confined faults and $1.1 \times 10^{-5}$ for unconfined faults.

(ii) The frequency of failure of the trip protection systems is predicted to $4.3 \times 10^{-6}$ for confined faults and $2.9 \times 10^{-7}$ for unconfined faults.

(iii) Taking operator errors into account, the upper bound to the frequency of sequences outside the design basis with the potential for core damage due to failure of the DHR systems is $3.3 \times 10^{-4}$ for confined faults and $1.6 \times 10^{-5}$ for unconfined faults.

A detailed further analysis was carried out to claim for natural circulation, and it showed that the effect is to reduce by a factor of about 2 the failure probability for the most frequent pressurised faults.

In conclusion, interesting results were obtained:

it was demonstrated that the weak system was the DHR one, and that in particular with the insertion of a new auxiliary boiler and with the installation of further system for feeding town water directly into the secondary the overall failure probability could be reduced to $1.4 \times 10^{-5}$ y$^{-1}$.

Moreover, it was demonstrated that the effect of installing a new protection system, completely separate and redundant would not significantly reduce the core damage frequency.

11 - FUTURE ACTIVITIES ON SITE

As previously said, the Station is now undergoing the total defueling of the core to the reprocessing plant.
Next activity will be the dismantling of the low radioactive parts of the plant (i.e. boilers) and the insulating of test facilities to reduce contamination and activity of components below established limits for unrestricted release. Other items will be conditioned for long-term deposit.

A program is under way to be settled in order to provide significant smear tests, specimens and calculations to obtain a full inventory of radioactivity of the primary circuit and samples for future testing.

It is also envisaged the conversion of the conventional part of the plant into a combined cycle with new gas turbogenerators and recovery of the exhaust gas heat to produce steam for the original main turbines. The plant siting and the steam cycle are strongly in favour of this solution.

REFERENCES


THE ISIS OPERATION: ROBOTICS REPAIR WORK ON THE CHINON A3 NATURAL URANIUM, CARBON DIOXIDE COOLED, GRAPHITE MODERATED REACTOR

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Abstract

After describing the upper internal support structures of the CHINON A3 reactor, the problems resulting from their degradation due to corrosion and to the difficulties of the ISIS operation are presented here.

The repair method is as follows: all tools and repair parts reach the working area by the feeding-pipes drilled through the 7 m thick concrete vessel surrounding the reactor core; the robots handle, into the reactor, the tool heads and the repair parts which are automatically positioned and welded around the corroded structure, thus restoring the support of measurement devices. The parts are either linked together or to the existing structure by means of 2 studs of 12 mm in diameter.

The different phases to sort out a problem are: in-core topography, reconforming of the full-scale mock-up with the repair area, learning on this mock-up and in-core repair.

The technical specificities of the robots used are the following: they have an 11 meter long, 0.22 meter across telescopic mast with joined arms reaching a radius of 2.7 m. Then the useful load is 70 daN and the repeatability 0.1 mm.

Different tool heads can be handled by the robot:
- telemeter and laser reconstruction: it allows to locate the in-core points and to materialize them on the mock-up by a laser crossed-beams locating technique,
- scouring: it cleans the corroded parts of the structures before welding,
- welding: it allows the parts handling and the carried studs welding,
- screwing,
- tensile test: carried out when the stud welds are defective.

A high level computerized control system is organized around a central unit which calculates the displacements of robots.