



ISOE NEWS

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International 2005 ISOE/EPRI ALARA Symposium

The January 2005 ISOE/EPRI ALARA Symposium was organized by the ISOE North American Technical Centre (NATC) in co-operation with Electric Power Research Institute (EPRI) at Ft. Lauderdale, Florida. There were 170 participants registered from 14 countries. Professionals from Canada, Europe, Japan, Korea, Russian Federation, and United States held presentations and shared their experiences.



The focus was on Industry Operational Experiences such as source term reduction, remote monitoring programmes, ultrasonic fuel cleaning, high level waste closures, and reactor coolant system equipment replacements. Some presentations were about utilities RP related trends and indicators. One day was dedicated to U.S. Industry Highlights presented by the representatives from EPRI, INPO and NEI.

Distinguished Papers

A team of peer judges from France, Japan, Korea, Romania, Russian Federation, Slovenia, and Spain selected three papers from this Symposium for presentation during the 2006 ETC Workshop in Essen Germany. Judging criteria included (1) consideration of the technical content, (2) the presentation effectiveness, (3) the applicability to other nuclear power plants, and (4) the timeliness of the subject. Selected papers are:

- "Pressurizer Heater Nozzle Replacement at San Onofre" by Kelli Gallion,
- "Ultrasonic Fuel Cleaning Pilot at Quad Cities" by Ken Ohr,
- "Browns Ferry Unit 1 Restart Source Term Reduction Initiatives" by John Underwood.

Congratulations to these three authors for their hard work and skilful presentations.

ALARA Performance Recognition



VC Summer Plant was awarded the NATC "World Class ALARA Performer" based on evaluation focused on its ALARA dose reduction plan, management and plant support. The evaluators were five peer radiation protection managers. The plant has implemented an effective source term reduction by new ion exchanger PRC-01 resins in past three refuelling outages (RFO). Collective dose is 0.47 manSv in 3-year rolling average (includes 2 RFO, reactor vessel head inspection, under vessel inspection, fuel pool re-rack). Last year's collective dose was 0.52 manSv.

U.S. Industry Highlights

NEI (Regulatory Support)

Ralph Andersen (NEI) presented the need for Radiation Protection Strategic Plan (RP2020) for the U.S. nuclear energy industry. In general, driving issues are the following: less frequent, more compact outages; increasing reliance on technology; mergers and acquisitions – culture and process integration; source term transients – complex chemistry; emerging significant work scope; degraded fuel performance; human resource and budget constraints; reliance on knowledge-based, rather than procedure-based, decision-making; and emerging new radiological protection standards.

EPRI (Technical Support)

Sean Bushart, EPRI Program Manager, explained EPRI Radiation Exposure Management Program. It consists of Radiation Protection and Radiation Field Control topics. The first part includes ALARA planning and material clearance; remote monitoring dosimetry and effective dose equivalent; worker safety and RP guidelines. The second topic of the program is dealing with cobalt replacement and Co-free hardfacings; surface conditioning – stabilized chrome electro-polishing; and UT fuel cleaning and LOMI decontamination.



EPRI Projects on RP Guidelines are dealing with methodologies needed for managing risks encountered in jobs performed in radiological environments and for optimizing worker safety. Remote monitoring technology guidance documents are in preparation. These solutions are needed for RP managers challenged with meeting strict outage schedules and reduced RP staff.

The impact of standards for clearance and exemption is a current objective. Overall goal is to develop a guidance for implementing these standards at NPPs.

ALARA related issues are dealing with site specific optimization of ALARA and RP technical resources. Industry focused project include assessments of cavity decontamination practices, new technology investigations, scaffolding management. ISOE/EPRI collaborative reports are on reactor vessel head inspections and replacements.

INPO (Ensuring Best Practices)

David Moss gave an INPO perspective on industry RP performance. Key elements are source term reduction in the first place; long-range goals and plans; strong line management ownership; maximum outage efficiency; and effective implementation of ALARA fundamentals.

2005 collective dose goals are based on utility input. The fact is that the median value is not moving down at a rate that will meet the 2005 goal. However, almost half of BWRs and PWRs are meeting goals and prove that it can be done. Assistance visits are focused on higher-dose plants, and working meetings are planned for generic issues. INPO also began collecting ALARA beneficial practices (available on web site). To gather solutions, a support is requested of RP managers and plant ALARA committees. It is important to implement a few good ideas.

Regulatory Agencies Highlights



The Symposium participant were informed about NRC findings related to radiation protection and source term events in 2004. A special meeting was organized after the Symposium by Jim Noggle (NRC Region I) to include first time regulatory benchmarking effort. Participants were from Canada, Korea, Spain, and United States. There is a plan to continue and expand for future meetings.

Pressurizer Heater Sleeve Replacement

(Kelli Gallion, San Onofre Nuclear Generating Station)

At San Onofre PWR (SONGS), testing of two Alloy 600 pressurizer heater sleeves revealed longitudinal indications in both sleeves and a circumferential indication in one sleeve. There were no through-wall cracks, and no external leakage was detected. One pressurizer heater sleeve was replaced during a prior refueling outage and replacement of all sleeves was scheduled for the next Unit 3 Refueling Outage (late 2006). Discovery of the indications during the October 2004 U3RFO (Cycle 13) precipitated a decision to replace all sleeves during this RFO with Alloy 690.

The repair plan required that the Alloy 600 sleeves be internally severed and that new "half-sleeves" be installed. Extensive machining and welding was required to replace 28 sleeves. A new primary system pressure boundary was established when the "half-sleeve" was installed. Job duration was 53 days with 18700 radiation work permit hours.



Heater Shielding

Major radiological issues included exposure and contamination control. Extensive engineering controls were used to reduce surface and airborne contamination levels. Much of the work was overhead and required that dosimetry be relocated to the workers' head. Work practices, remote monitoring, and temporary shielding on the sleeves, the surge line, and the work platform avoided about 0.17 manSv over the originally estimated 0.816. Total exposure for the job was 0.645 manSv. The highest individual exposure was 16.85 mSv. Lessons learned from similar work at Palo Verde were very helpful to SONGS.

Some Alloy 600 Facts

The Alloy 600 is subject to primary water stress corrosion cracking (PWSSC). It has typically a long incubation period (up to 27 years) and depends on: operating temperatures, heat treatment, cold work, and chemical environment. All Alloy 600 heats used in U.S. plants have been tested for PWSSC and failed (EPRI).

Locations where Alloy 600 PWSSC has occurred at PWRs are: reactor head nozzles, reactor vessel safe ends, hot leg nozzles, steam generator drains, pressurizer heater sleeves, and pressurizer water and vapor space instrument nozzles.

ISOE Presentation to the OECD/NEA CRPPH

The 63rd meeting of the NEA Committee on Radiation Protection and Public Health (CRPPH) took place at NEA Headquarters on 8-10 March 2005. The Chair of the ISOE Steering Group, Mr. Jean-Yves Gagnon (Gentilly-2 NPP, Canada), presented the accomplishments of the ISOE programme for 2004, and its plans for 2005. The key issues discussed included the migration of the ISOE databases to a web-based system, and the broad strategic review of the Programme and its work to identify and deliver products that are of value and use to its participants. Although the ISOE programme is operated under its own specific agreement among participants (the ISOE Terms and Conditions), the Committee appreciated consideration by ISOE of requests and suggestions. The Committee also recognised the value of the information and analyses provided by the ISOE Programme, and requested a more detailed reporting of findings and analyses at the next meeting of the CRPPH in 2006. The Committee thanked the Chair for the report, noted the work undertaken by ISOE, and approved the ISOE proposed programme of work for 2005.

Accident Source Terms for Light-Water NPPs

Historical overview

In 1962, the Atomic Energy Commission of U.S. issued Technical Information Document (TID) 148844, "Calculation of Distance Factors for Power and Test Reactors". In this document, a release of fission products from the core of a light-water reactor (LWR) into containment atmosphere ("source term") was postulated for the purpose of calculated off-site doses. In addition to site suitability, the regulatory applications of this source term (in conjunction with the dose calculation methodology) affect the design of a wide range of plant systems. The release is assumed to consist of 100% of the core inventory of noble gases and 50% of the iodines (half of which are assumed to deposit on interior surfaces). These values were based on experiments performed in the late 1950s involving heated irradiated UO₂ pellets.

Source term estimates under severe accident conditions became of great interest after the Three-Mile Island accident when it was observed that only relatively small amounts of iodine were released to the environment compared with the amount predicted to be released in licensing calculations. The Nuclear Regulatory Commission (NRC) began a research effort about 1981 to obtain a better understanding of fission-product transport and release mechanisms under severe accident conditions. In 1995, U.S. NRC published Regulatory Guide (NUREG) 1465, "Accident Source Terms for LWR NPPs". The primary objective of this report was to define a revised accident source term for regulatory application for future LWRs. For example, the table below is taken from NUREG-1465. In this table PWR releases into containment are presented for severe accident scenario.

Table 1- PWR Releases Into Containment (fractions of core inventory)

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5h	1.3h	2.0h	10h
Noble Gases	0.05	0.95		
Halogens	0.05	0.35	0.25	0.1
Alkali Metals	0.05	0.25	0.35	0.1
Tellurium Group	0.05	0.25	0.005	
Barium, Strontium		0.02	0.1	
Noble Metals		0.0025	0.0025	
Cerium group		0.0005	0.005	
Lanthanides		0.0002	0.005	

In this NRC report, it was suggested that iodine entering containment from the reactor coolant system is composed of at least 95% caesium iodide (CsI), with no more than 5% I₂ plus HI. Once within containment, soluble CsI will dissolve in water pools and plate out on wet surfaces in ionic form. If the pH is controlled to a level of 7 or greater, conversion to elemental iodine will be minimal. The fraction of iodine in organic chemical form was suggested to be 0.15%.

Phebus Tests - further international efforts for reducing nuclear and radiobiological risks

The Phebus international nuclear research safety programme started about 15 years ago in collaboration with the European Commission and Electricité de France, and with the other organizations from United States, Canada, Japan, Korea and Switzerland. The aim is to further improve the calculation software programmes used in safety assessments. The research carried out in this field will help optimise the action and procedures implemented to protect people and the environment in the event of a nuclear accident.

Over about 25 years, a large number of separate-effect tests have been performed, several computer codes developed and only a few integral test carried out. Such tests are very complex and need for heavy investments. The Phebus contributes at this latter level using an apparatus representing the main parts of a

PWR at a scale of about 1:5000. It is located at IRSN (Institute for radiation protection and nuclear safety) Chadarache, France. These tests have been defined by international partners of the programme; they differ in the nature of fuel burn-up and geometry, more or less oxidizing conditions, the level of fuel degradation and the conditions in the containment (e.g., acidity and the temperature of the water-filled sump).

The 5th test (FPT-3) of Phebus FP Programme took place on 18th of November 2004. The objective is to study phenomena that occur during serious reactor meltdown accident at PWR. Previous tests have provided new data on core meltdown mechanisms, on the radioactive products released and on their behaviour. The purpose of the last test is to transpose the results to the 1300 and 1400 MW reactors in France and to other types of PWR operating in Europe.

Scientific lessons learned from previous tests

The lessons learned from the Phebus FP programme can be divided into three categories: those that the specialists foresaw and which have been verified, those that were foreseen but inadequately, and those that were not foreseen.

■ Those that were foreseen ... and which have been verified

In this category, we include the main phenomenology of core degradation which was, globally, correctly predicted. As the tests progressed, these calculations turned out to be more and more accurate with respect to: the phenomena of clad oxidation and associated hydrogen production; material interactions; and release of volatile fission products. Regarding the primary system, the difference between calculated and experimentally-measured total retention of fission products was acceptable; the same thing can be said regarding the containment where calculation of the distribution and kinetics of aerosol deposits also agreed with experimental observations showing the deposits dominated by those in the sump and on cold surfaces. Other verified phenomena were fission product release at UO₂ dissolution, low release from the molten pool, and low total release of ruthenium.

■ Those that were foreseen ... but inadequately

Certain phenomena, though foreseen, were not well quantified where this affected both fuel-rod degradation and the primary circuit. For example, fuel liquefaction was not predicted at the temperatures observed but at temperatures about 400°K higher. The suppressed release of barium – a fission product that contributes in a major way to decay power – must also be cited where this release is large in separate-effects tests (from 50 to 90 %) but reduced to a lower level in the Phebus tests with its clad fuel rods. In the primary circuit, the measured deposition profile disagrees with the pre-test calculation: deposits are under-estimated in the hot section and over-estimated in the cold section. Furthermore, chemical form of caesium is not CsOH as it was predicted before. Still, it is not sure in what form caesium appears; there is a guess that molybdenum might interact with caesium.

■ Those that were not foreseen ...

Finally, unexpected phenomena have been brought to light during certain Phebus tests, the most noteworthy concerning iodine behaviour.

Firstly is the early detection of low but significant amount of iodine in a volatile form in the containment. The iodine arriving from the primary circuit was anticipated entirely in a condensed (aerosol) form where this would subsequently have deposited by sedimentation before volatilizing from the sump, all this occurring at a relatively slow rate. The unexpected, direct input of volatile iodine to the containment gas phase has important consequences in the case of loss of containment integrity during the first few days of an accident or in the case of resorting to filtered venting of the containment since emissions to the environment would essentially comprise gas-phase species (aerosols having largely deposited and the remainder being filtered by soil or the engineered venting systems).

Another surprising phenomenon was the suppression of iodine volatilization from the sump (in molecular I₂ form) due to the presence of silver released by melting of the silver-indium-cadmium rod, the alloy used in control rods of Westinghouse-type reactors. It is a »good news« but reinforces the previous mentioned

consequence. These two observations show that, in contrast with initial predictions, for a reactor equipped with this type of control rod and as a function of the accident sequence, the primary circuit can be the principal source of volatile iodine into the containment. Amount of organic iodide (CH₃I) is dominant versus molecular form (paints in the containment play a role here). It is an important consequence depending on scenarios because this dominant iodine form can not be filtered so efficiently as the molecular I₂ form.

Table 2 - Phebus FPT-1 PWR Releases Into Containment (fractions of core inventory)

Group	Element	PHEBUS FPT-1 HL break CL break		NUREG-1465 (comparison) Sum (using Table 1)	
Halogens	iodine	0.84	0.641	0.65	<u>gap release + early in-vessel + ex-vessel</u>
Alkali Metals	caesium	0.549	0.438	0.65	- for volatiles (halogens, alkali metals and
Tellurium group	tellurium	0.632	0.525	0.305	tellurium) after reactor vessel failure
Barium	barium	0.0077	0.0065	0.02	<u>gap release + early in-vessel</u>
Strontium	strontium		0.003	0.02	- for low volatile fractions, the ex-vessel
Noble Metals	molybdenum	0.303	0.23	0.0025	release is not taken into account.
	technetium	0.245	0.206	0.0025	These fractions are coming mostly from
	ruthenium	0.0063	0.005	0.0025	corium-concrete interactions that were
Cerium group	plutonium	0.00027	0.00023	0.0005	not simulated in Phebus.
	neptunium	0.0098	0.0083	0.0005	
Lanthanides	zirconium	0.00017	0.00015	0.0002	

The last - 5th test (FPT-3)

For the last test an early degradation of the fuel rod containing boron carbide was arranged. The power level, temperature, hydrogen production and some other parameters were recorded during the power excursion. During this experiment a rapid release of gaseous boron and carbon species took place. Among the carbon-containing gases foreseen to be produced, methane is expected to be susceptible for interaction with iodine to form organic iodide. The experiment should lead to an observation of a significant contribution of the containment sump to the generation of volatile iodine (due to the absence of silver and the acidic and evaporating conditions of the sump). The acidic sump media was first at 90° and then at 120°C. The fuel used in the test was BR3 24 GWd/tU together with B₄C control rod element. The coolant flow condition was poor steam leading to reducing condition. The experiment has been followed by »iodine chase«. Its time was limited due to decay of iodine I-131. Collection of the samples and subsequent counting has been performed up to February 2005.

The main characteristic of the FPT-3 test is due to the control rod composed of boron carbide, representing French 1300 and 1450 MW PWRs, boiling water reactors in Europe and certain Russian-design VVERs, whereas the control rod for previous tests were made from Ag-In-Cd to represent French 900 MW or Westinghouse-type PWRs. This difference may considerably influence the behaviour of iodine. Furthermore, the FPT-3 test shall make it possible to test and compare different catalytic recombiners that are designed to limit hydrogen risks inside the reactor containment in the case of a severe accident with core meltdown. Interpretation of all the tests will not be finished before 2009.

Conclusion

In conclusion, the lessons learned from Phebus Tests tests are very important: numerous advances have been made in the evaluation of emissions, and computer codes have been considerably improved by verification of fission products behaviour in accident conditions. The results of severe accidents R&D will also be used for the source term re-evaluation studies. These will improve estimation of releases into the environment on the basis of reasonably pessimistic scenarios and might influence emergency plans

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