BNL ALARA CENTER

ALARA NOTES

Number 9

Edited by

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From The Editors

There are several important developments to report on in this issue of ALARA Notes:

Third International Workshop on ALARA:

The agenda for the workshop is included in this volume of 'Notes'. Great interest is being shown in the workshop as evidenced by the fine speakers and the number of registered participants. Can you still register for the workshop? The answer is **Yes.** Although we prefer advance registration for effective planning, registration will remain open until May 9, 1994, the first day of the workshop. The hotel can accommodate a very large number of participants, but we urge you to act now since the number of rooms that are available at the lower conference room rate is limited. A workshop brochure with all the details is enclosed. A request for continuing education credits for attendance at this workshop has been made to the American Academy of Health Physics.

Upgrade of the ACE On-Line Information System:

It is becoming increasingly difficult for new users of ACE to acquire PC Anywhere III, the communications software necessary to log on to ACE. Moreover, the producer is about to withdraw all support from the product. We will therefore be changing to a different communication software called CLOSEUP. This software is readily available and, unlike PC Anywhere III, the new versions of the software are fully compatible with its older versions.

With the new software, you will find it significantly easier to install and to log on to ACE. We have taken this opportunity to make ACE much faster and simpler to use. ACE is now based on only four basic commands. The new user manual is only four pages long. All the instructions required to access and retrieve information through ACE are given in the manual. A copy of the new user manual and information on when the upgrade will go into effect is enclosed with this issue of 'Notes.'

New Information Available from ACEFAX

Our on-line fax-on-demand service has become popular beyond all our expectations. In the last few months there have been 1,050 calls on the system. The simplicity of picking up the handset of the fax machine and dialing for the required information seems to appeal to our users. In response to your needs, it was decided to add more information to ACEFAX. We have added over 300 documents from the NEWS database and about 50 from the JOBS database, and have included subject indices for several of our databases to make information searches easier. We will continue to transfer new databases to this system, add new subject indices, and maintain all databases up-to-date as far as our limited resources permit. A new list of documents on ACEFAX is included in this issue of 'Notes.'

Database on Dose Reduction Projects for Nuclear Power Plants:

We are about to publish volume 5 of NUREG/CR-4409, "Data Base on Dose Reduction Research Projects for Nuclear Power Plants," which contains information from our two oldest and most important databases. The document should soon be available through the National Technical Information Center. If you wish to access documents in the report even before it is published, you may obtain them through ACE or ACEFAX.

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Nuclear Plant Discharges Too Low to Detect

The Atomic Energy Control Board's (AECB) annual report was brought to the table in the House of Commons in June. In it, Canada's nuclear power reactors have been pronounced acceptably safe by the regulatory authority. However, the report notes there were 620 unusual events during 1992, of which 259 required a formal report to the AECB. The report also points out that of about 7,015 nuclear generating station workers who were exposed to radiation, none received a radiation dose greater than the legal limit of 30 mSv in a three-month period or 50 mSv per year. The average dose per worker was 2.5 mSv.

Doses from discharges at all power plants were found to be very low. They varied from 0.001 mSv/y to 0.019 mSv/y for people living near the stations. As of March 31, 1993, there were 3,743 licenses in effect for the use of radioisotopes in medicine, research, and industry. AECB inspectors carried out 3,297 inspections of licenses and identified 1,534 major infractions that could have affected radiation safety.

Inspectors undertook 106 investigations of unusual situations, issued three stop-work orders and initiated a number of prosecutions. Eleven of them were completed during 1992-93, of which nine were successful, with one case against a company pending.

Taken from, "Nuclear Plant Discharges Too Low To Detect," from AECB Reporter, Summer 1993, p.8. For detailed information and copies of the AECB Annual Report 1992-93, contact AECB (613/995-5894), Office of Public Information, P.O. Box 1046, Ottawa, Ontario, K1P 5S9, Canada.

Present and Future Safety Issues for Électricité de France (EdF)

EdF has 54 PWR nuclear units in operation with experience of more than 500 reactor-years. In terms of collective exposure, the 1992 average annual dose was 2.36 man-Sv. An ALARA approach is being implemented in all plants to reduce this to a value of around 1.5. A better value is expected for 1993.

During the past four years, there have been no accidents in French nuclear power plants. EdF realized that the best way to reach a high level of safety is to analyze all safety significant events in operation, to find their root causes, and to take the appropriate corrective measures.

In 1991, some corrosion on the vessel head penetrations was discovered. This led to large-scale actions--financially, technically, and in terms of human resources. The best experts were consulted.

In terms of management of the nuclear generation system, EdF's strategy has been costly, particularly due to the loss of availability. But this strategy--with management concentrating on defense-in-depth (e.g., inspection, leak detection, anti-ejection systems, repairs and replacements) remains an exemplary one from a safety viewpoint.

Damage has also been encountered on steam generator tube bundles, but EdF's surveillance policy has so far made it possible to avoid a tube rupture. All their plants are equipped with a nitrogen-16 leak detection system.

A first steam generator replacement took place in 1990, and a second was carried out at the end of 1993. EdF also reassessed its replacement strategy in 1992, with a view to increasing the quality of operation and achieving a high level of availability in units fitted with new equipment.

Taken from, "Present and Future Safety Issues for Electricité de France," by P. Tanguy, Nuclear Engineering International, pp. 54-55, Dec. 1993.

Listening to Reactor Pressure at the Boundaries for the Sounds of Cracks and Leaks

Acoustic emission methods were first considered for continuous surveillance of reactor pressure boundaries to detect cracking and/or coolant leakage in about 1965. A lengthy U.S. research and development program has provided the technology and supporting documentation to make their application a reality.

Acoustic emission (AE) enables the detection of energy released as a crack grows in a solid material or as a high-temperature, pressurized fluid leaks to atmosphere through a crack. This is accomplished by using highly sensitive, surface-mounted detectors. It can:

- Continuously survey location inaccessible during reactor operation.
- Detect cracking as it occurs, providing real-time indication of integrity degradation.
- Locate the site of cracking.
- Be interpreted to give an estimate of crack growth rate.

The research program to produce the technology was conceived in 1976 and continued for 15 years. Some major problems have been solved including coolant flow noise, sensor longevity, long-term service, identifying the AE signal and interpreting data. As the final step in evolving new technology and making it useful to industry, arrangements are being made to establish a commercial source to apply the technology on demand.

Taken from, "Listening to Reactor Pressure Boundaries for The Sounds of Cracks and Leaks," by P.H. Hutton, Nuclear Engineering International, pp. 38-40, Dec. 1993.

The Safety of French Pressurized Water Reactors: A Regulator's Perspective

Standardization is a key feature of France's pressurized water reactors. For the safety authority, this essential fact is the starting point for all in-depth safety assessments.

One of the most important safety issues is about reactor pressure vessel head penetration cracking, which were first observed in September 1991, and were found to be due to a stress corrosion phenomenon which affects the material (Alloy 600) from which the penetrations are made.

By the end of 1992, out of 18 reactors checked, 13 were shown to be affected, with 5% of the penetrations showing the anomaly. The same anomaly has since been observed on sister reactors built elsewhere in the world. The cracking, in its current form, does not compromise safety. Nonetheless, it requires certain measures to be taken:

- Extended inspections on the entire population of plants.
- Temporary repairs as soon as cracks are detected.
- Development of final solutions.

EdF has placed an order for 13 new vessel heads fitted with penetrations made from an improved material (Alloy 690) and is working to perfect a process to replace the affected penetrations.

The strategy put forward by EdF in response to the three measures was examined at the beginning of this year by DSIN (Direction de la Sûreté des Installations Nucléaires) and its technical support units. DSIN accepted it with the notable proviso that the temporary repair criteria be strengthened.

Taken from, "Facing up to the Future: The Safety of French Pressurized Water Reactors - A Regulator's Perspective," by A.C. Lacoste, Nuclear Engineering International, pp. 51-53, Dec. 1993.

Options For Leak Detection

Early detection of leaks from the reactor coolant pressure boundary is vital for reliable operation and the prevention of accidents. Acoustic detection methods can help.

Several methods can be used to detect leaks in the pressure boundary of nuclear reactors. The most important sources of leaks in both PWRs and BWRs are valves and pumps. Detection methods and success rates vary with reactor type.

In the U.S., the improved acoustic leak detection technology has been applied to both PWRs and BWRs. With this technology the pressure boundary can be monitored continuously by using acoustic emission sensors placed on wave guides in direct contact with reactor components. It has two great advantages: it reacts to a leak immediately, and it can be used to acquire quantitative information about the leak.

Industry practice has shown that changes of 0.5-1.0 gal/min in water flow rate can readily be detected in containment pumps by monitoring changes in flow rate, pump water level, or operating frequency of pumps. If pumps and tanks used to collect unidentified leakages and air cooler condensate were to include instrumentation to give an alarm when normal flow rates increased by 0.5-1.0 gal/min, they would detect increases in leakage in most cases.

Increased humidity in the atmosphere indicates that water vapor has been released to the containment. Dewpoint measurements can be used to monitor the humidity of the containment atmosphere--a one degree increase in dew point is well within the sensitivity range of available instruments. The monitoring of humidity levels is most useful as an alarm, or as an indirect indication of change to alert an operator to a potential problem. It is not necessary that all of the above-mentioned leakage detection methods or systems be used in a specific nuclear power plant. The final choice of detection methods should include enough different systems to ensure effective monitoring at times when one type of detection system may be ineffective or inoperable

Taken from, "Considering Options for Leak Detection," by D. Kupperman, Nuclear Engineering International, pp. 41-42, Dec. 1993.

Good Experience From Operation Of Replacement Steam Generators

Early results from the replacement steam generators supplied by Framatome indicate that the required improvements in performance, flexibility of operation, and better reliability and availability are being obtained.

The first replacement steam generators were designed with performance levels close to those of the units they replaced and Alloy 600 thermally treated tubing (600 TT) was still used. At the time, the main concern of the utilities was to try to eliminate the consequences of the extensive degradations encountered with Alloy 600 mill annealed tubing.

One of the later design changes is to increase reliability and availability. This requirement is met by using Alloy 690 TT, which is immune from primary water stress corrosion cracking and which provides high margins against secondary-side corrosion phenomena even in polluted environments. The first in-service inspection results show the limited operating experience has confirmed the excellent resistance of Alloy 690 TT to corrosion phenomena. No corrosion, fretting, or wear has been reported so far.

Early operating experience with the Framatome replacement steam generators has underlined the importance of ensuring secondary side cleanliness during the changeover. Particular care has been taken to avoid any ingress of impurities into the secondary side of the steam generator, when maintenance work or modifications are performed on the steam/water system during the replacement operation.

Taken from, "Good Experience from Operation of Replacement Steam Generators," by J.P. Billoue, Nuclear Engineering International, pp. 36-37, Dec. 1993.

Remote Handling Equipment Aids Bruce Nuclear Power Station

Specially designed tool carriers and work tables, positioned in shielding cabinets against each reactor face, are being used in the latest retubing of CANDU reactors.

The CANDU reactor is built for on-line refueling. The calandria encloses some 480 horizontal pressure tubes arranged in an approximately octagonal matrix. The fueling machines access both ends of these pressure tubes to insert and remove fuel bundles while the reactor is on-line.

The tubes must be replaced about every 15 years due to their elongation under the high neutron flux. Numet Engineering has designed and manufactured retubing tool carriers (RTCs) and work tables for the retubing of the CANDUs at Bruce, which can be operated either manually at the reactor face or remotely.

The RTCs are six-axis, gantry-type cranes which are mounted on the shielding cabinet. They are designed to transport and manipulate irradiated and non-irradiated reactor components, tools, and equipment, both within the shielding cabinet and between the reactor face and other equipment on the reactor vault floor. Combined with the vertical travel of the shielding cabinet, each RTC can access all fuel channels on its reactor face. They are normally operated locally by an operator or remotely via a local area network.

Work tables are multi-axis motorized lift tables, which drive along rails installed in the floor of the shielding. They support, position, insert, and remove tools and equipment during removal and installation phases of retubing. The work table can be also be operated either remotely or locally.

The remote and local control of both RTCs and work tables greatly reduce man-rem by enabling a tool automatically to locate itself with reactor components.

Taken from, "Remote Handling Equipment Aids Bruce," by G.S. Crawford, Nuclear Engineering International, pp. 33-34, Dec. 1993.

EPRI's Low Level Waste, Chemistry and Radiation Control Program

The Low Level Waste (LLW), Chemistry and Radiation Control Program is a new business element formed by EPRI's (Electric Power Research Institute) Nuclear Power Division to help member utilities manage low level radioactive waste, optimize water chemistry, and control radiation exposures as a means of reducing operation and maintenance costs. The primary audience of the program's activities will be utility radiation protection, chemistry, and LLW managers.

The future unavailability of LLW disposal sites, the rising incidence of stress corrosion cracking in reactor systems, and tightening exposure limits present triple challenges to utility staff, made more crucial by the need to reduce costs. This program provides the cost-effective technology essential to meet these challenges. In recent years, the program staff have worked closely with their utility counterparts to develop improved techniques, field test them at lead plants and assist other plants in utilizing the demonstrated technology. Over half the funds for this work are investments by member utilities over and above their EPRI membership dues. Typically, the base program budget funds the research and development phase, and the additional funding, either co-funding or tailored collaboration, underwrites first-of-a-kind demonstrations.

Taken from, "Introduction of LLW, Chemistry and Radiation Control Program," by Chris Wood; EPRI LLW, Chemistry, and Radiation Control News, p.1, Dec. 1993. For further information, contact Chris Wood, 415/855-2379, Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94303.

European Utility Requirements for the Next Generation of Nuclear Plant

European utilities are keeping an open mind on their options for the next generation of nuclear plant. Designs under consideration include PWRs and BWRs with both evolutionary and passive features. However, the main European initiative has been the EPR (European PWR), which is being developed jointly by NRI (Siemens/Framatome) in collaboration with EdF and a group of German utilities. The development of the design by NRI is being paralleled by a close collaboration between the German and French licensing authorities and the development of a set of requirements by the European utilities. The utilities are giving initial priority to generic requirements for PWRs and specific requirements for the EPR. In order to decouple the design process from variations between sites with respect to the distance of the nearest population to the reactor, and in the dose take-up pathways, EUR (European Atomic Energy Community) set radiological targets for normal operation in term of discharge rather than doses. The targets are set at levels that are stringent, but judged to be achievable on the basis of experience with the best current plants. The collective operator dose target is 0.7 man-Sv per GW-year. Design measures, choice of material and operating procedures are specified to ensure that operator doses are ALARA (As Low As Reasonably Achievable).

Taken from, "The European Utility Requirements for The Next Generation of Nuclear Plant," by J.A. Board, Nuclear Engineering International, pp. 37-40, Nov. 1993.

New Steam Generators For Doel 3

Belgium's first steam generator replacement was completed in September. Electrabel's Doel 3 plant received three new steam generators in an outage lasting 96 days.

The replacement was carried out by Belgatom as part of an outage that also included:

- Ten yearly inspection of the reactor vessel and internals.
- Modifications arising from the first 10 years safety reassessment of the plant, including recabling of electrical penetrations, upgrade of safety injection systems and containment spray piping, etc.
- Replacement of the process computer.
- Maintenance of pumps.
- Replacement of reactor coolant pump snubbers.

The outage was planned on-line hour-by-hour, and workers from Belgatom, Eletrabel, and subcontractor Siemens received a combined radiation dose of 1.96 Sv.

Installation of the new steam generators will also enable the plant to be uprated by 10% from its current 945 MWe.

Taken from, " New Steam Generators for Doel 3," Nuclear Engineering International, p.3, Nov. 1993.

Worker Protection during Contaminated Concrete and Steel Remediation

Concrete and steel are primary structural elements in nuclear facilities and industrial plants where hazardous radioactive materials and toxic chemicals are present. Although often covered with special protective coating, both the coatings and concrete of walls, floors, and structures become contaminated to some extent during their services life. This contamination can extend from fractions of an inch to several inches into the porous concrete, depending on the integrity of coatings and service conditions encountered.

Various chemical wash methods, wet and dry abrasive blasting techniques, and surface scarification processes have been applied for concrete and steel decontamination. During these decontamination operations, workers are required to wear burdensome protective clothing and respirators to protect themselves against harmful radioactive and chemical contamination. While in this protective clothing, worker's production rates dramatically decrease while fatigue, heat stress and risk potential increase. Due to the large volume of concrete and steel decontamination work to be completed at the Savannah River Site, the Westinghouse Savannah River Company (WSRC) implemented engineering controls that could reduce - or even eliminate - excessive worker protective gear while, at the same time, promote safe and aggressive production.

Taken from, "Engineering Considerations for the Needlegun with Local Exhaust," by S. Lefkowitz and G. Harris. For further information, contact George E. Harris, Pentek Inc., 1026 Fourth Ave., Corapolis, PA 15108-1659.

More Circumferential Cracking Indications Found in Ringhals 2 Upper Head

Circumferential defect indications were recently found at nearly every upper vessel head penetration of the Ringhals 2 PWR. By using a new type of miniature ultrasonic probe, the new type of circumferential indications have been found between the penetrations and the welds which join them to the head. They are probably small fabrication defects arising from localized lack of fusion. They are thought unlikely to be large enough to rose major integrity problems. Analysis is still underway.

Taken from, "More Circumferential Cracking Indications Found in Ringhals 2 Upper Head," Nuclear Engineering International, *p. 2, Nov. 1993.*

A Record-Breaking SG Replacement

Thirteen months of planning enabled replacement of steam generator at Beaznau to break all speed records.

Originally, the plan was to replace only the lower part of the steam generator. However, two studies carried out by Westinghouse and Sulzer, and evaluated by NOK (Nordostschweizerische Kraftwerk), came out strongly in favor of complete replacement with new steam generators, and NOK decided that this was a better option.

Contracts for the replacement were placed in August 1990. A project term was set up at the beginning of 1992. The group carried out 13 months of detailed planning and the plant outage began on 1 April with the replacement itself began on 12 April. Originally it was planned to take 46 days, but the tight scheduling enabled the replacement consortium to reduce time to 44 days. The primary and secondary circuits were handed back to NOK for pressure testing on 26 May.

Taken from, "Picture This: a Record-Breaking Replacement," Nuclear Engineering International, *pp. 22-24, Nov. 1993.*

Neutron Detectors

BTI Bubble Technology Industries developed two new neutron detectors devices. They are neutron detectors BD 100R O/F and BD 100R PND. They provide reusable, passive integrating dosimeter that allows instant, visible detection of neutron radiation. These detectors can be reset using a integral recompression device and can be reused for up to three months when reset daily.

They can be used under water with no effect on performance. They offer dose range two orders of magnitude more sensitive than any other commercial neutron dosimeter (<1 uSv) and has a flat energy response from about 200 keV to 14 MeV.

The BD 100R PND is now temperature compensated. The operation of the device is identical to the BD 100R O/F which has already gained acceptance in many areas of applications.

For further information, contact Dosimetry Service, 800/666-4552, 2501 Barrington Road, Hoffman Estates, IL 60195-7372.

[EDITORS' Note: Certain devices that appear to be particularly useful are sometimes mentioned for readers' information in ALARA Notes. However, this does not imply an endorsement of these products.]

An Electronic Personal Dosimeter System

Siemens developed a Electronic Personal Dosimeter System (EPD) recently. The EPD detects X-ray, beta, and gamma with a dose equivalent range of 0.1 mrem to 1,000 rem. It meets applicable DOE/EH-0027 and ANSI N13.11 requirements and is designed to be used for the dose of ordinary alarming dosimeters. It also features a unique personal computer linkage system which lets you track and record readings in your own database for immediate analysis and review.

For further information, contact Siemens Gammasonics Inc, 800/666-4552, Dosimeter Service, 2501 Barrington Rd, Hoffman Estates, IL 60195.

[EDITORS' Note: Certain devices that appear to be particularly useful are sometimes mentioned for readers' information in ALARA Notes. However, this does not imply an endorsement of these products.]

Scaffolding Management for Waste and Dose Minimization

Scaffolding is an integral part of most power plant outages and is required to provide access to areas that are not at convenient elevations for maintenance activities. The use of scaffolding in support modification and preventative maintenance activities for nuclear power plant can result in a number of burdens to site resources. Plants can choose either own scaffolding or rent when needed.

If it is rented, then health physics or decon is required to clean the scaffolding prior to returning it to the rental agency. If the scaffolding cannot be cleaned, then it must be bought and stored in a controlled area storage or disposed of as radwaste. This required capital expenditure to purchase the scaffolding, on top of the rental fees paid, when it would have been cheaper to have bought the scaffolding in the first place. Controlled area storage is not easily available at most plant, and the costs of radwaste disposal have skyrocketed.

If a plant decides to own scaffolding, the same inventory management, decontamination, storage, and radwaste issues must be dealt with. In addition, both rental and ownership options result in dose management issues that should not be overlooked.

In response to similar situations at numerous other nuclear plants, the concept of scaffolding management for contaminated materials was introduced by Quadrex in 1986. The program creates a pool of scaffolding which can be accessed by utilities. The 20-acre Recycle Center site in Oak Ridge, Tennessee, includes substantial areas dedicated to managing over 1.2 million pounds of scaffolding. Thirty-seven nuclear plants are regular users of or have used the scaffolding management program. The economies of scale associated with this large program allow for quick turnaround and customization to support station needs. Utilities can order only what they need, and the scaffolding arrives containerized and segregated in order to best support erection sequencing. Once the outage is over, materials are rapidly removed and loaded into the vans in no particular order and shipped to the vendor. On arrival, they are off-loaded, inspected, decontaminated, refurbished, coated, color-coded by length, and prepared for the next use. All knuckle-tightening bolts are manually run-out and thread-chased to ensure rapid erection. Poles are shipped on roller carts for ease of moving to erection areas.

At a recent outage, the utility tracked erection time and personnel exposure. Compared to in-house scaffolding management, this program allowed scaffolding erection in half the time and at 20 percent savings in dose.

Based on this program, the cost associated with health physics and security monitoring of controlled scaffold storage have been eliminated, and the capital costs associated with purchasing new scaffolding has been eliminated. In addition, radwaste volumes and personnel exposure were greatly reduced.

Taken from, "Scaffolding Management for Waste and Dose Minimization," by Frank J. Svetkovich and Michael S. McGough, Nuclear Plant Journal, September-October 1993, pp.54-56. For further information, contact Michael S. McGough, 904/373-6066, Quadrex Corp., 1940 NW 76th Place, Gainesville, FL 32606.

Reactor Vessel Stuck Stud Removal and Stud Hole Repair

Each removal and replacement of a reactor vessel (RV) head for a PWR means the removal and replacement of up to 60 reactor vessel studs. During the removing procedure a lot of factors can result in conditions that will cause a stud to seize in place and become impossible to withdraw by normal removal techniques.

Different tools and field methods have been utilized for removing seized RV studs. Of the methods available for removing seized studs, the boring and whirling unit, is capable of quick turnaround with assured success.

In late 1992, as part of the second refueling outage activities for Unit 1, Comanche Peak sought to remove one seized stud. At the same time, as part of the start-up activities for unit 2, the utility wanted to conduct stud hole video inspection and stud hole thread repair as well as remove a stud that had seized during hot functional testing.

The length of the task was an important consideration. The removal of the stuck stud in Unit 1 was scheduled to be completed between fuel off-load and fuel reload. The unit 2 tasks needed not to delay start-up schedules. Also, ALARA concerns required that the Unit 1 work be performed as quickly as possible. TU Electric designed and installed a reactor vessel plug to minimize radiation exposure. Nevertheless, radiation fields in the vicinity of the Unit 1 stuck stud were expected to be on the order of 1 mSv (100 mrem) per hour, since the reactor upper internals are stored on the cavity floor at the vessel flange level during refueling at Comanche Peak. For these reasons and because earlier attempts to remove the Unit 1 stuck stud using other methods had proved unsuccessful, the boring and whirling unit offered the best opportunity for achieving the task objectives. Work began inside the uncontaminated Unit 2 containment. Since the inspection and repair equipment was contaminated, it was necessary to take extra precautions in establishing a radiation control area (RCA) around the work area. Once the PCA was set up, equipment was moved into place.

When the work in Unit 2 was finished, the tooling and equipment were moved to Unit 1 to remove the other seized stud. Radiological surveys indicated radiation fields much higher than anticipated. Even with the reactor vessel plug in place and extensive temporary shielding between the reactor vessel and the upper internals, radiation fields ranged from 1.5 to 2 mSv (150 to 200 mrem) per hour in the cavity close to the stuck stud. Performing the work operations quickly became even more significant in order to minimize personnel exposures.

The boring and whirling unit proved to be a prudent means for achieving the task objectives. The stud removal for each unit was completed in just over one work shift. The sleeving operations were each completed in less than two work shifts. As far as radiation exposure is concerned, the Unit 1 stud removal resulted in less than half the total person-rem than would be expected using conventional techniques.

Time is an ever important factor in decisions regarding means and methods for accomplishing outage activities, both from a schedule and ALARA perspective. With stuck reactor vessel studs occurring more frequently, the boring and whirling unit provides a means of ensuring schedules are maintained and radiation exposure is kept to the lowest level.

Taken from, "Reactor Vessel Stuck Stud Removal and Stud Hole Repair," by Brain C. Elliott, Nuclear Plant Journal, September-October 1993, pp. 54-57. For further information, contact Brain Elliott, 804/385-2336, B&W Nuclear Technologies, 3315 Old Forest Road, P.O. Box 10935, Lynchburg, Virginia 24506-0935

Refurbishing Rather than Replacing Reactor Coolant Pump Motors

Reactor coolant pump motors are large, complex, expansive, and critical components, requiring about two years lead time to produce. Their failure can result in unplanned outages and significant replacement costs. Quadrex Motor Service Center provides an alternative to burial and replacement of failed contaminated motors, and to attempts to refurbish on-site in facilities ill-suited to such complex tasks.

Corrective or preventive maintenance work on radioactively contaminated electric motors done within the constrains of outage schedules at nuclear installations presents unique challenges. At substantial cost, some stations keep spare motors to use if failures occur. Without spares, unscheduled motor maintenance, a forced outage due to motor failure or replacement can result in an inconvenient and expensive rush. If spares are available, contaminated motor repairs can be carried out at a more leisurely pace, either at the plant site or at a motor repair shop licensed for radioactive materials. Alternatively, the motors can be decontaminated and shipped to an unlicensed facility for repair.

The benefits of using a dedicated motor service facility are demonstrated by recent experience with the Quadrex Motor Service Center, which has 9,500 square feet dedicated to RCP motor work and is segregated into the work arèas.

The building has two levels. One area, served by a 60-ton bridge crane, is high enough to remove the rotor during disassembly. The rest of the building is used for processing each of the motor parts as required for refurbishment. This area is served by a 15-ton bridge crane. The entire building is maintained at a negative pressure for contamination control and is segmented to minimize cross-contamination when working with more than one motor. Refurbishing North Anna's RCP motors is a five-step process: preparation; disassembly and inspection; repairs and modifications; reassembly and testing; and return to service.

Parts and components removed from the motor are cleaned and decontaminated following inspection. Except for special cases, to perform repairs, for example, decontamination is done for personnel exposure control or to reduce the potential for spread of contamination, and not to achieve a free-release condition. For the North Anna motor, wipe-down by hand was sufficient in most cases. The rotor was steam-cleaned to limit potential airborne contamination during the no-load test. Steam cleaning was followed by oven dry-out to assure complete moisture removal.

The most significant of the lessons learned was that stator destacking was required to rewind. The original plan was to decontaminate the stator core following winding removal, rework the core iron as necessary, and then rewind. Stator core destacking was a contingency task intended to be performed only if difficulties were encountered during decontamination or if the core could not pass a core loss test. Since replating and restacking, the stator core is expected to improve its performance, and since contamination levels should not be significantly different for the next motors, the refurbishment process for subsequent motors has been modified to proceed directly to destacking.

Taken from, "Refurbishing Rather than Replacing Reactor Coolant Pump Motors," by J.A. DeMarco, M.S. McGough, W.E. Mendez, and D.W. Heyer, Nuclear Engineering International, October 1993, pp. 38-40.

Service Water Restoration at North Anna

Most of the pipes of North Anna's service water system (SWS) are corroding, but they are encased in concrete or buried, making access difficult. Virginia Power has had to devise a way of remediating the piping before wall thickness falls below allowable limits.

Since the plant was built in the 1970s, the carbon steel piping of the SWS has corroded and suffered wall loss because the bacteria in the water promote aggressive corrosion of carbon steel materials. From 1982 to 1992, much of this piping was replaced with stainless steel, with good operational performance. However, the large-diameter uncoated piping continues to corrode. Compared different kinds of solution from both technical and cost standpoints, in-situ pipe refurbishment emerged as the only way.

The solution consisted of three measures:

1) Enhanced chemical treatment of the service water reservoir to control the bacteria in the system.

2) A comprehensive cleaning, repair, and coating program for 550m of concrete-encased 0.61m piping.

3) A replacement program for 91m of direct buried 0.61m piping.

The repair process for the concrete-encased sections consisted of four steps:

1) Initial grit blast cleaning of the pipe interior to remove corrosion products.

2) Engineering inspection of the cleaned surface to identify any areas required weld repair.

3) Weld repair.

4) Application of a two-coat 100% solids epoxy coating system. Approximately 240m of concrete-encased piping were restored in this manner.

When the project is completed in 1996, more than 640m of 0.61m diameter piping will have been coated internally.

Taken from, "Service Water System Restoration at North Anna," by Mark D. Sartain, Nuclear Engineering International, October 1993, pp. 34-36. The author is with Virginia Power, P.O. Box 2666, One James River Plaza, Richmond, VA 23261, USA.

Planning SG Replacement Pays Off for North Anna

Virginia power replaced the steam generator at North Anna unit 1 in 51 days and 30% under budget. They did it with careful planning and team work.

During the initial planning, Virginia Power had to decided to replace the bottom half encompassing the tube bundle. The utility also decided to rely on the two-cut strategy which was one cut on each reactor coolant loop at the elbow to nozzle joint. A like-for-like replacement steam generator was chosen as it would adequately support continued operation of the station and could be done under Nuclear Regulatory Commission requiremehts.

During the outage personal exposure needed to be kept as low as reasonably achievable (ALARA). Project exposure goals were set at 500 person-rem and the limit to the number of workers being accidentally contaminated was set at 110.

The project very successfully exceeded its goals. The total project exposure was 240 person-rem, less than 50% of the 500 person-rem target. Personal contamination events were held to 67, out of a target of 110.

The restart was accomplished without incident, and except for slightly longer than normal chemistry and calibration holds at intermediate power levels, proceeded normally to 100% power. The new steam generators are performing satisfactorily.

Taken from, "Planning Pays Off for North Anna," by Leslie L. Spain, Nuclear Engineering International, October 1993, pp. 30-33. The author is with Virginia Power, Innsbruck Technical Center, 5000 Dominion Boulevard, Glen Allen, VA 23060, USA.

Lasers Make Light Work of Sleeving

Laser welded sleeving was performed for the first time in the United States in April 1992, followed by a larger campaign in October 1992. These successful projects, as well as demonstrating the field hardiness of the laser equipment, showed that laser sleeving offers a degree of process control not found with other methods and produces welds that can be fully inspected by ultrasonics. A further application is expected in March 1994.

The use of laser welding for sleeve installation promises several advantages over current methods, which include the mechanical or hybrid expansion joint, tungsten inert gas welding, and explosive welding processes. Some of the advantages of laser welding are:

- It provides a hermetic seal.
- It addresses primary water stress corrosion cracking (PWSCC) as well as outer diameter stress corrosion cracking (ODSCC) mechanisms.
- It is insensitive to secondary side condition such as moisture and surface emissivity.
- It is tolerant of field variability in sleeve/tube fit-up.
- The focused application of energy results in smaller heat affected zone, lower heat input compared with current fusion welding methods, less distortion, and greater control of weld quality.
- The gradual sleeve hydraulic expansion process minimizes parent tube distortion.
- Repair welds can be made using the same process parameters as the initial weld.

The successful field implementation of laser welded sleeving has proven the feasibility of the process for repairing steam generator tubes and enhancing plant performance.

Taken from, "Lasers Make Light Work of Sleeving," by Bala R. Nair, Nuclear Engineering International, October 1993, pp. 26-30. The author is Manager, Advanced Technology Development, Nuclear Services Division, Westinghouse Electric Corporation, P.O. Box 158, Madison, PA 15663-0158.

Japanese Wrestle with Tube Problems

Japan's first PWR, Mihama 1, was commissioned in 1970 and soon suffered steam generator tubing problems. Since then, the industry has had to deal with a variety of tube degradation problem culminating in February 1991 with the country's first steam generator tube rupture. This was at Mihama 2, where Japan's first steam generator replacement project is now underway. Maintaining the integrity of steam generator tubing is today the most important factor in assuring plant reliability and safety in Japan. The technical standard set down in Japan requires that if a tube defect is found, the plant should not be operated. Therefore, all tubes are plugged or repaired by sleeving. In the early years, explosive plug and welding plug techniques were used for repair. Since 1981, mechanical plugging has been used in order to reduce the amount of radiation as well as working hours. From 1989, plugs of Alloy 690 have been used in order to reduce the sensitivity to stress corrosion cracking of mechanical plugs.

In 1980, sleeving instead of plugging was first applied in order to avoid a reduction of allowable plugging margin of steam generator. At first, the welding sleeve was used, and then the mechanical sleeve was employed, with which repair work is easier. Since 1984, brazing sleeves were used to repair the intergranular attack occurring in the tube support plate crevice. Laser welding of sleeves was put into use in 1989.

To maintain and enhance the integrity of steam generator tubing, national and joint research projects are being carried out by utilities and manufacturers. Their investigations cover the following area:

- IGA (intergranular attack) of Alloy 600.
- Chemical substances which affect the sensitivity to IGA.
- Concentration and precipitation behavior of trace impurities in water.
- The effect of various oxides on the electric potential of Alloy 600 in alkaline.
- Development of sleeving techniques.
- Verification of effectiveness of IGA prevention measures.
- Chemical cleaning of the steam generator secondary side.

Although various techniques have been applied, IGA has not yet been arrested completely, and remains the gravest threat to steam generators in Japan. That is why Japan's first steam generator replacement project is now underway.

Taken from, "Japanese Wrestle with Tube Problems," by Seiji Yashima, Nuclear Engineering International, pp. 23-26, October 1993. The author is Executive Director of Japan Power Engineering and Inspection Corp., Akasaka 1-chome, Minato-ku, Tokyo 107, Japan.

Paving the Way for Full System Decontamination with the Fuel In

The Indian Point 2 (IP2) full-system decontamination will be done with the fuel removed. Wastinghouse developed a program which involves the chemical decontamination of actual fuel assemblies in a specialized canister with the same dilute chemical solvent parameters as were employed in the full-RCS (reactor coolant system) qualification program.

To take account of current generation fuel and future generation fuel designs, one assembly of Vantage 5 and one assembly of Vantage-Plus type were exposed to CAN-DEREM solvents and one assembly each was exposed to LOMI solvents.

The two Vantage-Plus and twice-burned Vantage 5 assemblies were decontaminated and extensive TV visual and eddy current cladding oxide thickness inspections were performed before and after exposure to solvents with the same inspections planned after full cycle operation. Preliminary data shows no significant cladding corrosion performance differences between any of the assemblies. High-magnification TV visual examination of the grids, grid springs, and assembly nozzles, and hold downsprings show no adverse effects.

A number of decontamination process application anomalies were observed. These anomalies resulted in recommendations for further study.

Taken from, "Paving the Way for Full System Decon with the Fuel In," by R.S. Miller, P.E. Miller and D.R.. Peffer, *__EPRI Radiation Control News,__* *No. 18, August 1993, p.3. (Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94303)*

Developing the Reactor Operating Procedures for Full-System Decontamination

The objective of the EPRI reactor decontamination procedures development program was to develop the engineering evaluations, and procedures for the first U.S. full-RCS (reactor coolant system) chemical decontamination at IP2 (Indian Point 2).

The following information needs to be provided while preparing the procedures:

- Identify the various plant modifications and decontamination process system interfaces that are required to maintain the decontamination operating conditions and to preserve materials exposed to decontamination solvents.
- Identify operational and design requirement for off-normal NSSS (nuclear steam supply system) operations associate with the decontamination process, required interface between NSSS and the decon process system, temporary NSSS modifications, and the methodology for handling liquid waste volumes generated.

Westinghouse is also addressing the NSSS equipment and material concerns. The contractual work scope includes additional engineering evaluations, analysis, pre- and post-decontamination equipment testing, maintenance and inspection procedures, and the design and manufacture of a shield storage tank that will permit the removal of specific NSSS components prior to the start of the full-RCS chemical decontamination operation.

Taken from, "Developing the Reactor Operating Procedures," by T. Bengel, *__EPRI Radiation Control News,__* *No. 18, August 1993, p.3. (Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94303).*

Laying the Foundations for the Field Implementation of Full System Decontamination

Pacific Nuclear has turnkey responsibility for all engineering, equipment, construction, processing and waste processing services during the full reactor coolant system (RCS) decontamination at Indian Point 2 (IP2).

Installation of the decon equipment and tie-in to the plant RCA will require engineering and field implementation of several plant modifications.

The IP2 effort requires engineering and fabrication of decontamination process equipment, ion-exchange and resin storage capacity, a transfer module for chemical mix and injection and spent resin processing. A computerized control system for remote operation is also being designed. A pre-shipment test program will be conducted to simulate key decon and waste processing operations. Once the laydown areas and other site preparations are completed, the system will be installed and tested at IP2.

After the outage has started and fuel has been removed, a tie-in spool with a flow control valve in the RHR system will be installed. Flow will be diverted through this temporary spool piece to the decon system and then returned through the same spool piece.

A 6-day schedule for implementation of the decon process has been established. CAN-DEREM and AP (access permit) alternated during the decon process.

Spent resin will be dried and stored at a temporary on-site storage location. High integrity containers will be supplied with optional mixing blades for solidifying if required.

Plant restoration will be performed to support the plant outage schedule. Restoration will include replacement/removal of the temporary modifications. The decon process equipment will be demobilized after resin processing has been completed.

A final report will be prepared and issued describing the result of the decontamination process.

Taken from, "Laying the Foundations for the Field Implementation," by John Sheffield, EPRI Radiation Control News, No. 18, August 1993, p.2. (Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94303).

Preparing for Full System Decontamination at Indian Point 2: Utility Perspective

The first U.S. full-system decontamination is scheduled for 1995 at Con Ed's Indian Point 2 plant. Radiation fields at the plant have increased to a point where they are above the industry average. Numerous efforts to address the radiation exposure have not yielded the desired results.

In 1988, Con Ed, EPRI, ESEERCO, and nine other utilities began a qualification program for the chemical decontamination of the entire RCS of a Westinghouse PWR. This qualification program was successfully completed in 1991. The LOMI and CAN-DEREM processes, with an alkaline permanganate (AP) conditioning step, were qualified for use as long as the fuel was removed.

Before and after radiation surveys will be taken at about 50 points in the plant to determine the effectiveness of the decon. The final decontamination factor (DF) will simply be the average over the individual points. A DF between 4 and 6 will meet the contractual obligations.

In the qualification program, Westinghouse tried to estimate the recontamination rates based on industry experience and computer analysis. The result indicates a full-system decontamination would achieve a DF of 5 and that the benefit would last for 5 operating cycles, or at IP2, about 10 years. Potential exposure savings were then calculated for a range of plants. The exposure that could be avoided over 5 operating cycles ranged from 1,000 rem to 3,500 rem.

The measured recontamination rates of previously decontaminated subsystems are well below what had been projected in the full-system decontamination report. Two factors could be that IP2 has maintained the higher pH values recommended for the RCS and has worked to replace cobalt sources in the equipment.

If the recontamination rates were to continue at the low levels, the radiation levels may never return to the original levels and the exposure saved could be even greater than originally estimated. At minimum, this indicates that the recontamination rates were not overly optimistic. Data will be collected and presented as they become available.

Taken from, "Preparing for Full System Decontamination at IP2 - A Utility Perspective," by Jack Parry, EPRI Radiation Control News, No. 18, August 1993, p.2. (Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94303). The author is with Consolidated Edison Co. of New York, Indian Point NPP, Broadway and Bleakley Avenue, Buchanan, NY 10511, USA

The Road to Full RCS Decontamination

In controlling radiation exposures in the future, reducing out-of core radiation fields by chemical decontamination offers the best chance of continuing the downward trend in exposures.

Decontamination technology is cost-effective, based on applications of the LOMI decontamination process. Decon outages are shorter, manpower requirements are reduced, and work quality is improved. Utilities estimate savings of 13,000 person-rem from 27 applications.

Full-system decontamination offers several important advantages: lower background fields, more effective decontamination, and reduced recontamination rates. A full-system decontamination demonstration will be carried out at Indian Point 2 station in 1995 with the fuel removed, but significant advantages could be achieved in terms of critical path time and even lower recontamination rates with fuel in place.

Full-system decontamination for BWRs could be economically attractive for removing deposits in the lower parts of the BWR cores to aid inspection/repair or to remove radioactive material that could be redistributed to out-of-core areas on switching to HWC.

The LOMI process has already been tested successfully on BWR fuel, but far more radioactive material was removed than anticipated, thus reducing the probability of future applications. Development of the ELOMIX electrochemical ion exchange process is continuing and this could well change the economics of all decontamination applications in the future. Into the application phase, much of the emphasis in the past has been to show that no corrosion damage will occur from the decontamination solvents. The U.S. industry has successfully completed that phase with AP/CAN-DEREM and AP/LOMI qualified for PWRs and LOMI for BWRs. The emphasis is now on the practical aspects of application and improving cost effectiveness.

Taken from, "The Road to Full RCS Decontamination," by Chris Wood, EPRI Radiation Control News, No. 18, August 1993, p.1, For further information, contact Chris Wood, 415/855-2379, Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94303.

Chemical Decontamination Workshop

An EPRI workshop on chemical decontamination was held in Charlotte, North Carolina, on June 8-9, 1993.

Utilities face the triple challenges of increasing requirements for repair and inspection work, the need to control operations and maintenance, costs and more restrictive radiation exposure limits. The theme of the workshop was the role of decontamination technology in meeting these challenges.

Decontamination vendors reported on utility decontamination carried out since the last workshop in 1991. U.S. vendors reported the result of 30 decontamination applications worldwide, with a similar number being reported by European and Canadian organizations. This indicate a significant increase in decontamination activities in the past 24 months.

Utility experience and lessons learned were discussed in the next session. Generally high decontamination factors were reported, but the importance of thorough preplanning and flexibility in procedures to respond to unanticipated results were stressed, as at previous workshops. With increasing complexity in water chemistry, particularly in BWRs, the desirability of artifact testing before the decontamination of high hydrogen injection rates coupled with zinc injection in BWRs might produces the effectiveness of subsequent decontamination was discussed. The need to accommodate decontamination within shorter refueling outages was a growing challenge to the operators, as was leakage caused by access hole covers. Noteworthy by its absence from this session was any mention of corrosion problem resulting from decontamination; clearly the guidelines on reagent selection for susceptible systems have proved effective.

Recent technical developments have been primarily aimed at increasing the efficiency of decontamination reagents, reducing ion exchange resin requirements and improving analytical technique. New work showed no adverse corrosion effects and up to $30,000 savings in reagent and waste disposal costs for a typical recirculation piping decontamination.

Waste management issues are becoming a major impediment to increasing use of decontamination technology, and major progress has been made in PWR full-system decontamination technology.

A new topic for this series of workshops concerned harsh decontamination techniques for free release of replaced items and decommissioned components.

Workshop proceedings can be obtained from Christopher Wood at (415) 855-2379

Taken from, "Chemical Decontamination Workshop," EPRI Radiation Control News, p.1, July 1993.

Surrogate Video Tour

There is a new video "game" at Browns Ferry Nuclear Plant (BFN). The objective of the game is to avoid entering the RCA and by doing so maintain your exposure ALARA.

The name of this "game" is surrogate tour. The player sits at one of the work stations and selects the area he or she wishes to view using a computer mouse. The photographs are then sequentially displayed providing an illusion of "walk" through the plant.

The surrogate tour system is a very valuable gadget for saving dose to radworkers. Jobs can be "scoped" without entering the RCA and without any dose. The system has already proven it's value to plant management on numerous occasions during the recent U2C6 outage of BFN. This system has given outage management a view of plant equipment they may have never seen before, thus expediting their decisions which saves time, dose and money.

Taken from, "Surrogate Video Tour," Browns Ferry Nuclear Plant Radcon News, p.3, July 1993. For further information call Mike Scarboro (Editor) at 205/729-3400.

Temporary Shielding for RHR Piping

A temporary shielding package for the large bore RHR (residual heat removal) piping on top of the unit 3 Taurus of Browns Ferry Nuclear Plant has just been installed. The insulator craftsmen did an outstanding job installing over 1,000 lead wool blankets directly on the RHR piping. This shielding was installed to support recovery modifications starting with tours platform steel.

In addition, this shielding is expected to remain in place during recovery and will provide significant man-rem savings for other recovery tasks.

Taken from, "Temporary Shielding," Browns Ferry Nuclear Plant Radcon News, p.2, July 1993. For further information call Mike Scarboro (Editor) at 205/729-3400.

Browns Ferry Nuclear Plant Unit 2 Cycle 6 Chemical Decontamination Proven Success

By the end of July 1993, Browns Ferry Nuclear Plant (BFN) are at 82% of the overall site goal with 75% of the fiscal year expended. With unit 2 back on line, the exposure to fiscal year ratio is expected to even out by the end of the year. The focus of attention for exposure accrual has shift back on unit 3.

Chemical decontamination project was accomplished during the U2C6 outage in support of the AS Low As Reasonably Achievable (ALARA) concept. There were those who were quite skeptical of the successfulness of chemical decontamination to appreciably lower work area radiation dose rates. However, as proven by post outage exposure data, this effort successfully lowered the radiation exposure required to complete outage work in the drywell and reactor water clean-up (RWCU) rooms within the rector building.

The RWCU system decontamination was initiated with chemical injection on 2/23/93 and ended with chemical clean up on 2/26/93. A three-step process known as LOMI-AP-LOMI (i.e., Low Oxidation State Metal Ion and Alkaline Permangate Oxidation) was utilized. The entire RWCU system flow path was decontaminated with the exception of the demineralizes.

The results:

- 10 pounds of metal oxides removed
- 11 Curies of radioactivity removed
- average contact decontamination factor (DF)=20
- average general area DF=6
- average pre-decontamination contact does rate=304 mr/hr
- average post-decontamination contact does rate=41 mr/hr
- average pre-decontamination general area dose rate=78 mr/hr
- average post-decontamination general area dose rate=17 mr/hr

The Reactor Recirculation (RECIRC) system and Residual Heat Removal (RHR) system piping within the drywell was decontaminated. The process initiated with chemical injection on 2/28/93 and ended with chemical clean up on 3/4/93. A three-step LOMI-AP-LOMI process was utilized.

The results were

- 41 pound of metal oxides removed
- 108 Curies of radioactivity removed
- average contact DF=70 (RECIRC); 29 (RHR)
- average general area DF=8 (RECIRC & RHR)
- average RECIRC pre-decon contact dose rate=388 mr/hr
- average RHR pre-decon contact dose rate=192 mr/hr
- average RECIRC post-decon contact dose rate=18 mr/hr
- average RHR post-decon contact dose rate=42 mr/hr
- average pre-decon general area dose rate=24 mr/hr
- average post-decon general area dose rate=7 mr/hr

The total estimated person-rem savings as a result of the project was 595. The estimated project cost required to complete the decontamination was $2.2 million. Based on this data, the net estimated benefit for completion of the project in support of U2C6 outage tasks was $6.0 million.

Taken from, "Unit 2 Cycle 6 Chemical Decontamination Proven a Success," Browns Ferry Nuclear Plant Racon News, p.1, July 1993. For further information call Mike Scarboro (Editor) at 205/729-3400.

Automating Pump Nozzle Inspection

The problems caused by the complex weld geometry of BWR axial pumps have been overcome in designing equipment for completing an ultrasonic scan where manual inspection was required before, reducing the radiation dose received by the operators. Because of the complex geometrical and ultrasonic boundary condition of the nozzle-to-vessel welds of axial pumps, a suitable manipulator unit with an associated control system was developed.

The manipulator has three degrees of the freedom: circumferential and axial movements as well as probe rotation. It travels on a hinged annular rail around the nozzle and is equipped with a movable vertical boom, the upper end of which carries the swivel probe on a Carden joint. The rail on the skirt for inspecting the bottom closure is used for transporting the manipulator system from one pump nozzle to the next, minimizing manual work inside the skirt itself.

The three-axis control system is designed to perform both horizontal and vertical meanders at a programmable constant inspection density with optimum parameters for ultrasonic inspection, i.e., with the beam axis perpendicular to the most probable direction of a crack. Scanning is controlled to avoid obstacles like weldments and instrument lines in the scanning area automatically. The probe can also be removed manually for alignment and calibration and for any additional analyses of indications that may be necessary. They are matched to the specific geometrical boundary conditions, such as radius of curvature, wall thickness, and scan area. To avoid damaging the thermal insulation, the couplant water is collected and discharged by pump.

The ultrasonic instrument is an "Impulse 1" with a dynamic range of approximately 80 dB to measure the maximum echo amplitude in the programmable gate range together with the corresponding time of flight.

Taken from, "Automating Pump Nozzle Inspection," Nuclear Engineering International, p.39, July 1993.

Automating Inspection of Vessel Penetrations in French PWRs

A variety of ultrasonic heads for detecting defects can be installed on MIPAC, an integrated inspection system built by Intercontrole in France to inspect vessel penetrations. The desire to reduce the exposure of operators to radiation, combined with the large number of inspections that needed to be made, motivated efforts to automate the procedure. This team has carried out vessel operation with television equipment. The mobile base of a television manipulator used for that work was adapted for MIPAC by connecting a motorized telescopic arm and a transducer carrier. MIPAC enters under the closure head through a door in the biological shield, so no manipulation of the closure head itself is required. Since the manipulator is operated remotely, manual work under the closure head is avoided. The centering plate installed under the closure head stand in each French plant is used for centering MIPAC automatically. A remote power unit supplies the motors, actuators and water pumps for the ultrasonic coupling, which makes the equipment autonomous.

The control and command system, which is linked with a camera, allows the ultrasonic head to move into position quickly. A laser target helps to center it in the penetration to be inspected. The operators stand in the annular gallery 20 to 50m from the manipulator.

Specially focused transducers are used. The different models applied, which are fitted into removable examination heads, operate under water. MIPAC's ultrasonic equipment is suitable for detecting both circumferential and axial defects, as well as for controlled inspection of the penetration weld geometry. The miniaturization needed was obtained by using a special arrangement of transducers and curved mirrors. Scanning is achieved by combining rotational and translational movements of the head.

Data processing is achieved by using an ultrasonic system currently used for inspecting French nuclear vessels. The software manages acquisition and digitization of the complete ultrasonic signals and analyses them on a Hewlett-Packard workstation. The diffraction echoes from the tips of the cracks allow the defect geometry to be determined. Processed results are printed out in color.

The equipment was used at a rate of less than 50 min per penetration, a complete vessel head inspection can be carried out in a week. The cumulative personal dose is around 10 mSv or 1 rem for the inspection of 65 penetrations.

Equipment is being developed to inspect penetrations without having to remove their thermal sleeves. Eddy current will provide detection, and defects will be characterized by ultrasound, from the 3-mm gap between the thermal sleeve and the penetration.

Taken from, "Head to Head: Automating Inspection of Vessel Penetrations in French PWRs," Nuclear Engineering International, p.36, July 1993.

Japan is Nearly Reaching the Limits of Automated Inspection

In their continuing efforts to prevent component failures in LWRs and to reduce the time taken to check for defects, Japanese engineers are automating many of the techniques previously applied manually. The limitations on the use of automated equipment arise because of:

- Narrow Space: The space available to set up a general purpose automated ultrasonic system is general restricted by support structure and obstructions.
- Time required for fitting: Application of an automated ultrasonic system does not always reduce the expose of its operators to radiation, because of the time it takes to set up and later remove, a system and to adjust it *in situ*.
- Large development costs: Developing an automated ultrasonic system takes a long time and requires procedure qualification tests before application on-site, making the process expensive. An advanced eddy current detection system for fatigue cracks in steam generator tubes.
- Difficulty in attracting investment: It is difficult to quantify the economic advantages attributeable to automated systems through improved inspection accuracy and reduced radiation exposure.
- Reluctance to replace manual ultrasonic methods: Manual ultrasonic systems are simple to use and are usually sufficiently accurate, so their efficacy may lead to the simple view that it is not necessary to achieve better accuracy by using an automated ultrasonic system.

Nevertheless, despite these limitations, there are some new inspection technologies, which do seem to promise genuine advantages:

- An automated ultrasonic inspection system for the Advanced BWR bottom head.
- A general purpose ultrasonic inspection system for butt and fillet welded joints.
- An ultrasonic data recording system for three-dimensional components.
- A magnetic crawler-type inspection robot.
- An advanced eddy current detection system for fatigue cracks in steam generator tubes.
- Multi-channel digital eddy current instruments for steam generator tubes.

Taken from, "Are We Reaching the Limits of Automated Inspection in Japan?" Nuclear Engineering International, p.34, July 1993.

Combining AI and NDE to Aid Pipework Repair and Inspection Decision

Computer tomography, ultrasonic holography, and electromagnetic acoustic transducer technology (NDE) have been combined with software using AI principles to create a system that focuses inspection work on the parts of plant pipework most vulnerable to cracking. Ultimately, the system could enable staff without specialist knowledge to take decision about repairs.

First, a database was used to predict the locations where defects are likely to occur. In parallel with the database, a defect identification program which identifies the defect by matching with the database was developed to identify the types of defect that occurred and to evaluate quickly and automatically the degree of their potential harmfulness to safety if the plant were to continue operation. To achieve these aims, it was decided to use the development tools of an expert system using an artificial intelligence (AI) language for the main defect identification program.

If a defect is detected, or if the program for evaluating crack initiation probability indicates the likely occurrence of a defect, a sectional image of the area concerned could be inspected by using the specially developed X-ray computer tomography (CT) scanner.

Digital ultrasonic holographic equipment was developed to obtain data for the defect identification program. It provides information about the shape, dimensions and inclination of the defect from the three-dimensional image of a defect detected by in-service inspection.

The electromagnetic acoustic transducer serves a similar purpose to the ultrasonic holographic unit.

Taken from, "Combining AL and NDE to Aid Pipework Repair," Nuclear Engineering International, p. 31, July 1993.

Replacement of Separator Shroud Bolts

Due to the IGSCC (intergranular stress corrosion cracking) found on two separator shroud bolts, Chin-Shan NPS replaced those bolts during its EOC-13 outage. Hydrolyzing was adopted as the major dose reduction measure by eliminating predicted radioactive crud which already existed since reactor startup. A DF (decontamination factor) of 30 was obtained with this 5000 psi, 3 gpm decon process. The replacement cost totaled man-hours and 2.88 mMan-Sv.

The requirements for dose reduction were fully understood and prepared for by the persons delegated to carry out the task. The hydrolazer was the key to reducing this unique close-body exposure condition. Contact dose rate before decontamination was approximately 60 mSv/h, but reduced sharply to 5-6 mSv/h, and furthermore to 2mSv/h on some spots after application. The operating platform also indicated a 0.3 mSv/h reduction of the field dose rate. Related measures taken to achieve dose reduction are:

- Underwater operated hydrolyzing with proper water removed crud successfully. Based on previous experiences, the effectiveness could be enhanced with 8,000-12,000 psi, if available.
- We estimate a saving of half of the exposure time, called close-body exposure, by adjusting the platform on proper positions for chain block operation.
- Underwater TV monitored the hanging-up process, bolt knobs were rarely blocked preventing unnecessary exposure.
- Continuous tap water was applied to prevent airborne contamination from drying the bolts.
- Double PC (protective clothing) with hood kept workers clean.

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