**EFFECT OF DECONTAMINATION ON AGING PROCESSES
AND CONSIDERATIONS FOR LIFE EXTENSION***

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EFFECT OF DECONTAMINATION ON AGING PROCESSES AND CONSIDERATIONS FOR LIFE EXTENSION

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ABSTRACT

The basis for a recently initiated program on the chemical decontamination of nuclear reactor components and the possible impact of decontamination on extended-life service is described. The incentives for extending plant life beyond the present 40-year limit are discussed, and the possible aging degradation processes that may be accentuated in extended-life service are described. Chemical decontamination processes for nuclear plant primary systems are summarized with respect to their corrosive effects on structural alloys, particularly those in the aged condition. Available experience with chemical cleaning processes for the secondary side of PWR steam generators is also briefly considered. Overall, no severe materials corrosion problems have been found that would preclude the use of these chemical processes, but concerns have been raised in several areas, particularly with respect to corrosion-related problems that may develop during extended service.

INTRODUCTION

This paper describes the basis for a recently initiated program on the decontamination of nuclear reactor components and the possible impact of decontamination on extended-life service. This program is being developed in response to the ever-increasing use of chemical decontamination processes in nuclear power plants to reduce worker exposures during repairs and maintenance, and to a growing interest in extending the life of existing nuclear plants beyond their present 40-year licensing period. Concerns about the possible adverse effects of repeated decontaminations on the remaining service life of aged components must be addressed before operating licenses for extended life can be renewed. Although a wide variety of nuclear power plant components are subject to aging effects of one kind or another, the present program is specifically concerned with metallurgical and chemical effects in components that are directly exposed to decontamination solutions, i.e., selected pressure boundary components and reactor internals. In addition, chemical cleaning of steam generators to remove sludge and reduce denting, though not strictly a decontamination process, is also included.

PLANT LIFE EXTENSION AND COMPONENT AGING

The duration of a license for a nuclear power plant was set at a maximum of 40 years by the Atomic Energy Act of 1954, with possible renewal at the end of that time period.¹ Several studies²⁻⁴ to evaluate the technical and economic merits of extending service lives beyond this 40-year limit have found that such life extensions are technically feasible and that the economic incentives are considerable. A 1982 evaluation performed for the Electric Power Research Institute (EPRI) concluded that such life extension could be cost-effective even at a plant refurbishment cost of \$300 million to \$1 billion.³ Robert B. Minogue, former Director of the Office of Nuclear Research of the U. S. Nuclear Regulatory Commission (USNRC), has noted that present 40-year operating licenses for nuclear power plants will begin to expire in the year 2001, and by 2015 approximately 85 percent of the licenses granted through 1985 will have expired.⁵ It is therefore not surprising that the USNRC has already begun to seriously examine the safety implications associated with nuclear power plant life extension.⁶

The major concern with respect to the possible licensing of nuclear plants for operation beyond 40 years is that of component aging. In terms of metallurgical effects, such aging can take several forms, including low-temperature sensitization of stainless steels, long-term embrittlement of cast duplex stainless steels, and radiation-induced embrittlement of reactor internal components. Each of these aging phenomena will be discussed briefly.

It has been recognized only relatively recently that sensitization of stainless steels can occur after prolonged exposures (of the order of years) to temperatures of 400°C (750°F) or less.^{7,8} This process, known as low-temperature sensitization, generally requires an initial brief exposure to higher temperatures (600 to 800°C or 1110 to 1470°F) to nucleate the carbide particles that subsequently grow at lower temperatures. Just as with conventional sensitization, this precipitation process depletes the grain-boundary regions of Cr and leaves them more susceptible to intergranular corrosion. Low-temperature sensitization is of concern in nuclear plant coolant piping and reactor internals operating at temperatures of 250 to 288°C (480 to 550°F), particularly under conditions of extended-life operation.

Another aging phenomenon, sometimes known as 475°C embrittlement, can occur in ferritic and cast duplex stainless steels. When these steels are exposed for extended periods of time to temperatures in the range from 300 to 500°C, the Cr-rich α' phase precipitates within the ferrite, resulting in severe embrittlement. It now appears that a similar series of time-dependent precipitation phenomena may occur at LWR operating temperatures in the ferrite phase of cast duplex stainless steels such as CF-3 and CF-8, which are used to fabricate nuclear plant primary coolant piping as well as valve and pump casings.⁹ Again, these very slow aging processes are of greatest concern under conditions of extended-life operation.

The embrittlement of reactor pressure vessels and internal components due to neutron irradiation may also be considered to be an aging phenomenon of sorts. This embrittlement process is generally characterized in terms of an increase in the transition temperature between brittle behavior observed at lower temperatures and ductile fracture at higher temperatures. This transition temperature increases with increasing neutron fluence (and therefore service time), and it has also been found to be strongly influenced by the presence of certain impurities such as Cu and Ni in the steel. Control of these residual element levels has greatly reduced the embrittlement problem in the newer reactor pressure vessels, but older vessels will require careful monitoring, particularly if extended-life operations are attempted.

CHEMICAL DECONTAMINATION

During the operation of a nuclear power plant (BWR or PWR), oxide corrosion films form on the surfaces of piping, pumps, valves, steam generator tubing, and other components in contact with the primary coolant. Some of these corrosion products dissolve or erode in the circulating coolant, are deposited on the fuel cladding, and become activated. The principal radionuclides formed by this activation process are ^{58}Co , ^{60}Co , ^{55}Fe , ^{59}Fe , ^{54}Mn , ^{51}Cr , ^{63}Ni , and ^{65}Zn . These radionuclides, in turn, are released and carried out of the core region by the coolant and are incorporated into the oxide films on the surfaces of the primary coolant system components. The result is a steadily increasing radiation field on these components.

This radiation field is of principal concern during reactor outages for maintenance or repair, since it limits the working time of personnel in the region. In fact, radiation exposure represents a specific cost, commonly quoted as \$5000 to \$7000 per man-rem,^{10,11} that electrical utilities must bear during maintenance and repair procedures. Decontamination of primary loop components to reduce radiation fields and lower the man-rem exposure is therefore of vital economic interest to utilities. For example, based upon experiences with the Dresden 3 and Quad Cities 1 and 2 reactors, Commonwealth Edison Company reports average net savings of \$1.25-million from decontamination of recirculation piping preparatory to performing in-service inspection and repair work.¹² Reduction of the radiation exposure of nuclear plant personnel through effective decontamination is also consistent with the NRC objective of maintaining such exposure as low as reasonably achievable (ALARA).

The results of a survey of U. S. experience with the chemical decontamination of commercial nuclear power reactors are summarized in Table I. This tabulation is not necessarily exhaustive, but it does include a large majority of the chemical decontaminations that were performed in this country through the end of 1986. Most of the decontaminations were carried out on BWR primary coolant recirculation systems (PCRS) and reactor water clean-up systems (RWCU). In PWRs, the components most commonly decontaminated have been steam generator channel heads. Almost all of these decontaminations were carried out immediately prior to maintenance

Table I. Summary of Experience with Chemical Decontamination of Commercial Nuclear Power Reactors in the United States.

Date	Plant	Process	Components Decontaminated*	Years Since Startup
Boiling Water Reactors				
1964	Shippingport	Turco 4502/4521	PCRS	7
6/76	Dresden 1	NS-1 (conc.)	Primary system test loop	16
4/77	Peach Bottom 3	NS-1 (conc.)	RWCS	3
9/77	Peach Bottom 2	NS-1 (conc.)	RWCS	3
10/79	Vermont Yankee	CAN-DECON	RWCS	7
3/80	Brunswick 2	CAN-DECON	RWCS	4
3/81	Nine Mile Point	CAN-DECON	5 primary recirc. pumps	4
4/81	Brunswick 1	CAN-DECON	RWCS	4
10/81	Vermont Yankee	CAN-DECON	RWCS	9
12/81	Brunswick 1	CAN-DECON	aux. steam system	5
4/82	Peach Bottom 2	CAN-DECON	RWCS	8
4/82	Nine Mile Point	CAN-DECON	2 primary recirc. pumps	12
5-8/82	Nine Mile Point	CAN-DECON	PCRS	12
12/82	Hatch 1	CAN-DECON	RWCS	7
4/83	Peach Bottom 3	CAN-DECON	RWCS	8
3-4/83	Vermont Yankee	CAN-DECON	PCRS	10
10/83	Quad Cities 2	CAN-DECON	PCRS	11
10/83	Dresden 3	CAN-DECON	PCRS and RWCS	12
1/84	Pilgrim	NS-1 (dilute)	PCRS	11
3/84	Monticello	LOMI	PCRS and RWCS	13
4/84	Quad Cities 1	CAN-DECON	RWCS	12
4/84	Brunswick 2	Citrox	RWCS	8
5/84	Millstone 1	CAN-DECON	PCRS and RWCS	14
7/84	Peach Bottom 2	CAN-DECON	PCRS	10
7-3/84	Pilgrim	NS-1 (dilute)	PCRS and RWCS	12

Table I. (Cont'd.)

Date	Plant	Process	Components Decontaminated*	Years Since Startup
9/84	Dresden 1	NS-1 (conc.)	PCRS	24
10/84	Dresden 2	NS-1 (dilute)	PCRS	14
12/84	Cooper	Citrox (dilute)	PCRS	10
1984	Quad Cities 2	CAN-DECON	RWCS	12
10/85	Vermont Yankee	LOMI	PCRS and RWCS	13
11/85	Dresden 3	LOMI	PCRS and RWCS	14
1/86	Quad Cities 2	LOMI	PCRS	13
5/86	Quad Cities 1	CAN-DECON/LOMI	2 irradi. fuel assemblies	14
6/86	Oyster Creek	LOMI	PCRS	17
7/86	Brown's Ferry 1,2,&3	7 processes compared	6 RWCS pumps	9-12
12/86	Dresden 2	LOMI	PCRS and RWCS (hot leg)	16

Pressurized Water Reactors

1976	Surry	NS-1 (conc.)	Evaporator	3
9/82	Surry 2	CAN-DECON	Channel head (cold leg)	9
11/82	Surry 2	LOMI	Channel head (hot leg)	9
4/83	Ginna	CAN-DECON	Channel heads	13
8/83	Millstone 2	OZOX	Channel heads	8
8/84	Palisades	CAN-DECON	Channel heads	13
5/85	Millstone 2	Citrox A	Channel heads	10
1985	Indian Point 3	LOMI	Channel heads	9
1/86	Connecticut Yankee	CAN-DECON	Channel heads	18
6-8/86	H. B. Robinson	LOMI	Isolated primary pump	15
1986	Millstone 2	OZOX-A	Channel heads	11

* PCRS = Primary Coolant Recirculation System
RWCS = Reactor Water Cleanup System

or repair work to reduce worker exposure. The principal decontamination processes were CAN-DECON, NS-1, and, in more recent years, LOMI. It is apparent from the table that chemical decontamination is becoming an increasingly common practice in this country.

During a nuclear reactor decontamination, the chemical solutions are likely to come in contact with a number of different structural alloys. The more important alloys are listed in Table II along with examples of components in which they are utilized. Although core decontaminations with the fuel in place are not yet being routinely performed in this country, they have been performed in Canada (Douglas Point and Pickering Candu Reactors) and the U. K. (Winfrith SGHWR), and are of growing interest in the U. S. For this reason, Zircaloy cladding material is included in Table II. The high-strength Ni-base Inconel Alloys 718, 625, and X-750, which are used for core structural applications, are included for the same reason.

It should be noted that several of the alloys listed in Table II are susceptible to the long-term metallurgical aging effects previously described. For example, the austenitic stainless steels (SS) and, to a lesser extent, Incoloy 600 are subject to low-temperature sensitization, although Type 316 NG SS has been specifically developed to resist sensitization. Possible aging embrittlement of the cast duplex stainless steels CF-8 and CF-3 has already been discussed, as has the neutron irradiation embrittlement of the carbon steel pressure-vessel materials.

CORROSIVE EFFECTS OF DECONTAMINATION PROCESSES

Table I indicates that the most widely used decontamination processes in this country are CAN-DECON and, more recently, LOMI, and these two processes are of principal interest in the present program. Both processes have been extensively studied for possible corrosive effects on the reactor structural materials with which they come in contact during use. The results of these studies are summarized in this paper.

CAN-DECON Process

The CAN-DECON process is a dilute chemical decontamination process developed by Atomic Energy of Canada Ltd. (AECL) during the early 1970's.¹³ The process was first used on a commercial basis in the United States to decontaminate the RWCS of the Vermont Yankee BWR in late 1979. Since that time, more than 20 decontaminations of BWR's and PWR's have been carried out in this country with CAN-DECON.

The principal active chemical reagent used in the CAN-DECON process was originally Nutek L-106, sold by Nuclear Technology Corp. of Amston, CN.¹⁴ Since 1977, when London Nuclear Decontamination Ltd. (now LN Technologies) became the licensee of the AECL for the application of the process, the chemically similar reagent LND-101A has been utilized. This reagent consists of a mixture of organic acids (including oxalic acid) to

Table II. Nuclear Reactor Structural Alloys Likely to be Exposed to Decontaminant Solutions During Chemical Decontamination.

Alloy	Application
Austenitic Stainless Steels	
Type 304	Primary coolant piping
Type 316 NG	Primary coolant piping
Types 308 and 309	Corrosion-resistant cladding on pressure vessels and piping
Cast Duplex Stainless Steels	
CF-8	Primary coolant piping;
CF-3	valve and pump casings
Carbon and Low-alloy Steels	
A106	Primary coolant piping
SA333, Grade B	Primary coolant piping
A516, Grade 70	Primary coolant piping
A508, Grade 2	Reactor pressure vessel
A533, Grade B	Reactor pressure vessel
A302, Grade B	Steam generator shells
A570, Grade 40	Steam generator support plates
Nickel-base Alloys	
Inconel 600	PWR steam generator tubing
Inconel 718	Core structural components
Inconel 625	Core structural components
Inconel X-750	Core structural components
Special Purpose Alloys	
Zircaloy-2 and -4	Nuclear fuel cladding
Haynes Stellite 6B	Control element drives
Haynes Stellite 6	Weld-deposited hard facing on pumps and valves

dissolve the radioactive oxide films present on the components and a chelant to prevent redeposition of dissolved radioactive metal ions. The oxide films formed on PWR steam generator components typically contain more Cr (as much as 40 wt. %) than is present in typical BWR corrosion films.¹⁵ Chromium is present as Cr^{3+} ions, which are insoluble in the LND-101A reagent and its modifications. It is therefore necessary to utilize an alkaline permanganate solution (LND-104) to oxidize the Cr^{3+} ions to the soluble Cr^{6+} state. In decontaminating PWR components, therefore, the reducing LND-101A treatments are typically preceded by or are alternated with the oxidizing LND-104 treatments.

The CAN-DECON LND-101A solution is typically injected into the circulating reactor coolant in sufficient quantity to achieve a diluted concentration of 0.1 wt. % in solution. A side stream of circulating solution is passed through a filter and cation resin columns to remove particulate and dissolved materials. The normal decontamination temperature is 120°C (248°F), but decontaminations have been carried out successfully at temperatures from approximately 80 to 135°C.¹⁶

Concerns about the possible detrimental effects of LND-101A on LWR structural alloys were first raised in early 1982 in conjunction with the decontamination of the Nine Mile Point 1 BWR primary coolant recirculation system (PCRS).¹⁷ This decontamination was carried out in order to reduce worker exposure during repairs to safe end-to-pipe welds in the 28-in. diameter coolant lines, which developed through-wall leaks in the heat-affected zones of the welds. A Task Force formed to study the Nine Mile Point 1 failures also considered the question of whether the CAN-DECON decontamination performed immediately prior to component removal contributed to the observed cracking. The Task Force report, issued in September 1982, concluded that "...the 1982 total recirculation system decontamination does not appear to have exacerbated the observed cracking."¹⁷ Its only clear effect, insofar as the cracking was concerned, was "...the increase in sensitivity for inspection of cracks which already existed in the system."¹⁷ However, some of the piping welds were heavily electropolished after decontamination to further reduce radiation levels, and this electropolishing treatment removed some of the original inner surface where intergranular attack (IGA) induced by the decontamination process might have occurred. Furthermore, the failed piping was removed immediately after decontamination and therefore never saw actual reactor service. The Task Force report cautioned that longer term exposures after decontamination might reveal effects related to subsequent crack initiation.¹⁸

Laboratory tests of the CAN-DECON LND-101A reagent and studies of its possible contribution to the corrosion of reactor structural materials were conducted in 1984 by the General Electric Co. under EPRI sponsorship.¹⁹ Three LWR structural materials, namely sensitized Type 304 stainless steel, Inconel 600, and SA 533 Grade B pressure-vessel low-alloy steel, were evaluated in laboratory tests. Decontamination processing for 48 h at 120°C in a 0.1 wt. % solution of LND-101A produced relatively high corrosion rates of about 0.18 mm/day in the low alloy steel, but the rates for the other two

alloys were of the order of 10^{-3} mm/day. However, the decontamination process was found to have produced IGA to a depth of about 0.2 mm (2 to 3 grain diameters) in the Type 304 stainless steel specimens and 0.09 mm (~ 3 grain diameters) in the Inconel 600 specimen. In addition, surface pitting was observed in these latter two materials. Subsequent constant extension rate tensile (CERT) tests at 288°C (550°F) in water containing 0.2 ppm dissolved oxygen indicated an increased susceptibility to stress corrosion cracking (SCC) in the sensitized Type 304 stainless steel. In the case of field-decontaminated Type 304 stainless steel BWR pipe welds, shallow IGA was observed in the heat affected zone, and dense shallow pits were found in one specimen. However, neither of these features adversely affected the IGSCC resistance of these specimens as measured by CERT tests.

As a follow-up to the General Electric study, London Nuclear conducted a comprehensive review of general, galvanic, crevice, and pitting corrosion data and SCC test results for the CAN-DECON process obtained by them and others over a number of years.^{16,20-23} The alloys evaluated included several carbon steels, austenitic and ferritic stainless steels, nickel-base alloys, Zircaloy-2, and Stellite 6. The LND-101A reagent concentrations in these tests ranged from 0.05 to 0.3 wt. %; solution temperatures ranged from 25 to 135°C; and decontamination times ranged from 6 to 550 h. This study concluded that the LND-101A solution does not, in general, produce any undesirable corrosion effects in the evaluated alloys. The highest corrosion rates were observed in the carbon steels, with a maximum rate of about 0.19 mm/day. However, this rate could be reduced by about a factor of 10 by the addition of suitable corrosion inhibitors. Non-sensitized Type 304 stainless steel and Inconel 600 showed no increase susceptibility to IGSCC. In the sensitized conditions, these two alloys showed a significant susceptibility to IGSCC after a 500-h treatment when the solution contained no ferric ions. The addition of ferric ions at concentrations of ~ 50 to 80 ppm in the decontaminant solution eliminated the susceptibility to IGA and IGSCC. The enhanced IGA and IGSCC susceptibility previously observed in the General Electric studies was therefore attributed to a lack of ferric ions in solution, a condition that would not normally be present during actual reactor decontamination, except possibly for a brief period near the beginning of the decontamination process. For this reason London Nuclear has introduced a reformulated version of LND-101A, known as Rem-E-D, which contains ferric citrate additions to ensure a sufficiently high concentration of ferric ions during the initial stages of decontamination. In addition, London Nuclear concluded that the presence of oxalic acid in the LND-101A reagent can lead to IGA of certain alloys under severe conditions. Therefore, reagent LND-107, in which the oxalic acid has been removed, was also introduced as an alternative modification of the original LND-101A.¹³

LOMI Process

The LOMI (low oxidation-state metal ion) decontamination process²⁴ was developed by the Central Electricity Generating Board in Great Britain, and is currently licensed to several firms for use in the United States. The decontaminating reagent consists of low concentrations of picolinic acid and

vanadium (II) formate plus a chelating agent, and the process works by converting the ferric ions in the oxide layer on the contaminated component surfaces to the more soluble ferrous ions, with the V^{2+} ions serving as the reductant. For PWR oxide films, which typically have a higher Cr concentration, prior alkaline or nitric acid permanganate (AP or NP) oxidation treatments are commonly used. The LOMI reagent is injected into the circulating reactor coolant as with the CAN-DECON process, and a side stream of circulating coolant plus decontamination solution is again passed through a filter and cation exchange resins columns to regenerate the chelant. The normal decontamination temperature of 80 to 90°C (176 to 194°F) for the LOMI process is somewhat lower than the decontamination temperature for the CAN-DECON process.

Both General Electric Company¹⁹ and Pacific Northwest Laboratories (PNL)²⁵ have conducted studies on the corrosive effects of the LOMI process on LWK structural materials. The General Electric study found the general corrosion rates of SA533 Grade B carbon steel, Type 304 stainless steel, and Inconel 600 to be approximately one order of magnitude lower in the LOMI reagent than in the CAN-DECON reagent, and no IGA or increased susceptibility to SCC was observed. PNL similarly found that both lightly and heavily sensitized Type 304 stainless steel and Inconel 600 specimens exhibited low general corrosion rates in the LOMI solution. The Inconel 600 specimens exhibited some shallow pitting after AP, AP/NP, and AP/NP/LOMI treatments, but the pits repassivated in subsequent post-filming. No IGA or increased susceptibility to SCC was observed under any of the test conditions for either alloy studied by PNL.

Chemical Cleaning of Steam Generators

In the past nine years, the Electric Power Research Institute Steam Generators Owners Group has developed solutions for the chemical cleaning of the secondary side of PWR steam generators to remove Fe and Cu corrosion produce sludge deposits. Because of the large volume of material to be removed, the solutions tend to be more concentrated than those used for primary system decontaminations, and the formulations that have been developed for the Fe and Cu removal steps are somewhat different.²⁶⁻²⁸ The principal active ingredient in both cases is ammonium EDTA (ethylene-diaminetetraacetic acid), which is present at a concentration of 10 wt. % in the Fe solvent and 5 wt. % in the Cu solvent. The Fe solvent also contains 1 wt. % hydrazine to reduct Fe^{3+} to Fe^{2+} , which is more easily complexed by EDTA. The two solvents are typically used sequentially, with interspersed rinses, and the application temperatures are 88 to 96°C (190 to 205°F) for the Fe solvent and 32 to 43°C (90 to 110°F) for the Cu solvent. In addition, a stronger solvent containing 20 wt. % ammonium EDTA has been developed for dissolving sludge from crevices.

These concentrated solutions tend to be more corrosive than the dilute solutions used in primary system decontamination, and a series of corrosion tests was carried out during their development.^{26,29,30} This series consisted mostly of general corrosion tests, but limited crevice, galvanic, pitting, and

U-bend tests were also conducted. Essentially all of the alloys found in PWR steam generators were tested; weldments and sensitized test specimens were included. Corrosion coupons were also included as a part of the Millstone 2 steam generator cleaning conducted in 1985.²⁷ For the most part, the general corrosion rates of the carbon and low-alloy steels have typically been found to be less than 0.13 mm (5 mils) per application, and serious galvanic, crevice, or pitting effects were generally not observed. In view of the limited number of steam generator cleanings expected over the life of a PWR, these rates appear to be acceptable. However, unacceptably high localized corrosion rates of carbon and low-alloy steels were observed in some sample weldments, particularly when galvanically coupled to Inconel 600. This effect appears to be influenced most strongly by variations in solvent corrosion inhibitor and weld configuration. Additional site-specific weld region corrosion testing and qualification was recommended before using the chemical cleaning process at a specific plant.

POSSIBLE INTERACTIONS BETWEEN DECONTAMINATION AND AGING

The aging effects discussed above are produced by metallurgical changes that occur throughout the volume of the material, and one would not expect these bulk aging effects to be significantly influenced by decontamination, which is a surface process. However, the microstructural changes produced at the surface of a component by an aging process such as low-temperature sensitization may affect the response of that component to decontamination. Decontamination surface effects can, in turn, greatly influence subsequent corrosion and crack initiation and growth at the component surface. Thus, the principal interaction between decontamination and metallurgical aging is likely to be the effect of aging on decontamination response and the subsequent effect of any surface changes produced by decontamination on corrosion and crack initiation and growth.

An early concern with decontamination was that it might cause IGA and possibly an increased susceptibility to SCC, particularly in sensitized material. This concern has been shown to have some basis in fact for the CAN-DECON process under certain severe conditions, as discussed above. However, the follow-up tests performed by London Nuclear and others provide persuasive evidence that IGA and SCC would not be expected under normal decontamination conditions, and, in addition, the CAN-DECON reagents have been reformulated to further reduce the likelihood of such problems.

The relatively high general corrosion rates observed for carbon and low-alloy steels in the CAN-DECON LND 101A reagent are also of concern. Corrosion rates approaching 0.2 mm/day have been observed, and repeated decontaminations, as during extended-life operation, could lead to corrosion wastages in excess of the corrosion allowances for some components. The presence of cracks through the weld-deposited austenitic stainless steel cladding of the reactor pressure vessel would locally expose the underlying low-alloy steel to these solutions. Corrosion inhibitors have been found effective in reducing corrosion rates, but the sulfur compounds commonly

found in these inhibitors may cause pitting in the austenitic stainless steels. In the case of the LOMI reagent, the general corrosion rates of the carbon and low-alloy steels are about an order of magnitude lower, but excessive corrosion may still occur in PWR decontaminations during the NP oxidizing pretreatment. For both BWR and PWR decontaminations, very little data are available on possible corrosion effects on the high-strength Ni-base alloys used for core structural applications.

The effects of decontamination on aged cast duplex stainless steels have not been evaluated experimentally. The concern here is that, under certain conditions, decontamination might produce microscopically sharp surface irregularities through such localized corrosion processes as IGA, galvanic corrosion at the boundaries between the ferrite and austenite grains, or preferential attack of one of the two phases. These irregularities could, in turn, contribute to reduced impact strength by acting as stress risers. This problem would be aggravated under conditions of extended-life service, since repeated decontaminations would accentuate the surface irregularities and the duplex alloys would have more time to embrittle because of aging effects.

Finally, additional data are needed on the corrosive effects of the more concentrated solutions used to clean the secondary sides of steam generators. The localized corrosion of carbon and low-alloy steel weldments has been discussed above, and the contribution of this wastage to the possible fatigue cracking of secondary feedwater lines owing to cyclic thermal stresses produced by stratified flows must be considered.³¹ In addition, further studies are needed on possible IGA and SCC effects, particularly on alloys that have been sensitized by welding or long-term aging.

SUMMARY

In summary, the considerable body of data on and experience with the chemical decontamination of nuclear reactor primary systems indicate no severe materials corrosion problems. However, concerns remain in several areas, including (1) the high corrosion rates of carbon and low-alloy steels in the CAN-DECON and possibly in the LOMI NP processes, and the effects of corrosion inhibitors on the pitting of stainless steels; (2) the possible non-uniform corrosion of the cast duplex stainless steels that results in the introduction of microscopic surface features which may accentuate possible aging embrittlement in these materials; and (3) the very limited data available on the high-strength Ni-base alloys used for core structural applications and on materials subjected to long-term aging. Additional data are needed on possible localized corrosion and SCC effects associated with the chemical cleaning of the secondary side of PWR steam generators, particularly on aged and sensitized steam-generator alloys and weldments. These concerns and data needs will be addressed in the experimental portion of the present program.

REFERENCES

1. A. K. Banerjee, C. F. Bergeron, and W. B. Dodson, "An Approach to Nuclear Power Plant Life Extension," Nucl. Saf. 27 (1986), 385-390.
2. C. A. Negin, L. A. Goudarzi, L. U. Kenworthy, and M. E. Lapidès, "Planning Study and Economic Feasibility for Extended Life Operation of Light Water Reactor Plants," Report TPS 78-788 (CONF-790923), Electric Power Research Institute (September 1979).
3. C. A. Negin, R. S. Walker, and S. B. Shantzis, "Extended Life Operation of Light Water Reactors: Economic and Technological Review," EPRI NP-2418, Electric Power Research Institute (June 1982).
4. Northern States Power Company, "BWR Pilot Plant Life Extension Study at the Monticello Plant: Phase 1," EPRI NP-5181M, Electric Power Research Institute (May 1987).
5. R. B. Minogue, "Trends in Nuclear Safety Research--Looking Ahead to the 1990's," Proc. Thirteenth Water Reactor Safety Research Information Meeting, NUREG/CP-0072, Vol. 1 (February 1986), pp. 1-8.
6. U. S. Nuclear Regulatory Commission, "Nuclear Plant Aging Research (NPAR) Program Plan," NUREG-1144 (July 1985).
7. M. J. Povich, "Low Temperature Sensitization of Type 304 Stainless Steel," Corrosion 34 (1978), 60-65.
8. M. J. Povich and P. Rao, "Low Temperature Sensitization of Welded Type 304 Stainless Steel," Corrosion 34 (1978), 269-275.
9. O. K. Chopra and H. M. Chung, "Aging Degradation of Cast Stainless Steel," Proc. Fourteenth Water Reactor Safety Research Information Meeting, NUREG/CP-0082, Vol. 2 (February 1987), pp. 119-142.
10. R. A. Shaw, "Getting at the Source: Reducing Radiation Fields," Nucl. Technol. 44, (1979), 97.
11. J. E. LeSurr and G. D. Weyamn, "Cost-effectiveness of Dilute Chemical Decontamination," Proc. 1981 ANS Winter Mtg., San Francisco, CA, Nov. 29-Dec. 3, 1981; Trans. Am. Nucl. Soc. 39 (1981), 865.
12. C. J. Wood, "Decontamination Helps Control Nuclear-maintenance Costs," Power 129 (January 1985), 29-33.

13. R. Knox, "How Independent Tests Have Eliminated Can-Decon Corrosion Concerns," Nucl. Eng. Int. 32 (March 1987), 48-51.
14. J. F. Remark, "Plant Decontamination Methods Review," EPRI NP-1168, Electric Power Research Institute (March 1981).
15. J. L. Smee, "Decontaminating LWRs with Can-Decon Dilute Processes," Nucl. Eng. Int. 30 (January 1985), 29-33.
16. J. P. Michalko, P. J. Bonnici, and J. L. Smee, "Compilation of Corrosion Data on CAN-DECON™, Volume 1: General, Galvanic, Crevice, and Pitting Corrosion Data from CANDU and BWR Tests," EPRI NP-4222, Vol. 1, Electric Power Research Institute (October 1985).
17. Anon., "Decon Cracking Fears are Allayed," Nucl. Eng. Int. 29 (October 1984), 11.
18. Anon., "Does Can-Decon Encourage Pipe Cracking?," Nucl. Eng. Int. 30 (February 1985), 13.
19. M. T. Wang, "Corrosion Evaluation of Two Processes for Chemical Decontamination of BWR Structural Materials," EPRI NP-4356, Electric Power Research Institute (December 1985).
20. J. P. Michalko and J. L. Smee, "Compilation of Corrosion Data on CAN-DECON™, Volume 2: Influence of CAN-DECON on Stress Corrosion Cracking--Summary of Testing, 1975-1983," EPRI NP-4222, Vol. 2, Electric Power Research Institute (October 1985).
21. P. J. King and B. D. Warr, "Compilation of Corrosion Data on CAN-DECON™, Volume 3: Influence of CAN-DECON Process on Stress Corrosion Cracking--1984 Constant-Extension-Rate Tests," EPRI NP-4222, Vol. 3, Electric Power Research Institute (January 1987).
22. J. L. Smee and V. C. Turner, "Compilation of Corrosion Data on CAN-DECON™, Volume 4: General, Galvanic, Crevice, Pitting, and Stress Corrosion Data From PWR Tests and Applications," EPRI NP-4222, Vol. 4, Electric Power Research Institute (January 1987).
23. J. L. Smee and V. C. Turner, "Compilation of Corrosion Data on CAN-DECON™, Volume 5: Influence of the CAN-DECON™ Process on Stress Corrosion Cracking--Summary of Testing 1984-1985," EPRI NP-4222, Vol. 5, Electric Power Research Institute (January 1987).

24. A. Cruickshank, "Developing Techniques for Decontamination," Nucl. Eng. Int. 28 (November 1983), 41-44.
25. R. L. Clark and R. L. McDowell, "Corrosion Testing of LOMI Decontamination Reagents," EPRI NP-3940, Electric Power Research Institute (March 1985).
26. P. C. Hildebrandt, J. E. Nestell, R. C. Trench, and R. D. Varrin, "Weld Region Corrosion During Chemical Cleaning of PWR Steam Generators, Vol. 1: Overview and Discussion," EPRI NP-5267, Vol. 1, Electric Power Research Institute (July 1987).
27. N. R. Stolzenberg and R. C. Thomas, "Employing a Chemical Method for Tubesheet Sludge Removal," Nucl Eng. Int. 32 (January 1987), 39-41.
28. Anon., "Removing Magnetite from the Oconee Steam Generators," Nucl. Eng. Int. 32 (March 1987), 51-52.
29. E. L. Capener, G. R. Egan, and T. J. Feiersen, "PWR Steam Generator Chemical-Cleaning Data Base," EPRI NP-3477, Vols 1 & 2, Electric Power Research Institute (April 1984).
30. J. L. Barna et al., "Weld Region Corrosion During Chemical Cleaning of PWR Steam Generators, Vol. 2: Tests and Analyses," EPRI NP-5267, Vol. 2, Electric Power Research Institute (July 1987).
31. Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, Report of the USNRC PWR Pipe Crack Study Group, NUREG-0691 (September 1980).