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Materials Reliability Program: Assessment of Westinghouse and Combustion Engineering Nickel-Based Alloy Orphan Locations (MRP-274)

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Materials Reliability Program: Assessment of Westinghouse and Combustion Engineering Nickel-Based Alloy Orphan Locations (MRP-274)

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REPORT SUMMARY

The aging degradation of austenitic nickel-based alloys and their associated weld materials represents a major cause of component degradation in pressurized water reactor (PWR) systems. This report evaluates the potential for aging degradation of components inside the reactor vessel that are fabricated from austenitic nickel-based alloys and their associated weld metals and that have not been addressed in detail in other documents and programs.

Background

Aging degradation of nickel-base alloys has impacted the reliability of critical safety systems and required costly repairs and replacements. It has also attracted the attention of the U.S. Nuclear Regulatory Commission (NRC) and a number of groups and organizations in the nuclear industry. This report investigates components fabricated from austenitic nickel-based alloys and associated weld metals that have not been addressed in detail in other documents and projects. These components are called the nickel-based alloys orphans locations. The report also discusses some potential avenues for mitigating their aging degradation.

Objective

To assess the aging degradation of Westinghouse and Combustion Engineering nickel-based alloy orphan locations.

Approach

The project team first identified the affected nickel-based alloy components for the Westinghouse and Combustion Engineering PWR designs. A number of these components, such as vessel penetrations and nozzles, have already been addressed in other documents and programs, and were therefore excluded from consideration. The project team evaluated the remaining components for the potential occurrence of a number of different types of aging-related degradation mechanisms, including irradiation effects, primary water stress corrosion cracking (PWSCC), low temperature crack propagation (LTCP), and wear. For those aging degradation effects of concern, the project team proposed some potential approaches to mitigation.

Results

The expected time to PWSCC initiation was calculated using the effective degradation years (EDY) concept, which compares degradation at a particular temperature to degradation at a reference temperature of 600°F (315.6°C). Operating experience has shown that in PWR primary water, crack initiation in Alloy 600 materials can occur after 8 EDY. Using an assumed vessel inlet temperature of 560°F (293°C), the first stress corrosion cracking would not be expected

until after approximately 43 effective full-power years. The time to initiation would be significantly longer for a plant with a lower vessel inlet temperature.

Wear is only expected for the clevis inserts in the Westinghouse design and for the core stabilizing lug shims in the Combustion Engineering design. Plants monitor these locations on a regular basis as part of their American Society of Mechanical Engineers (ASME) Code Section XI inservice examinations.

The project team evaluated irradiation effects, including irradiation-assisted stress corrosion cracking, irradiation-induced stress relaxation, irradiation embrittlement, and irradiation-induced void swelling for each of the components. Since all of these components are in locations remote from the reactor core, the cumulative end-of-license neutron fluence will be too low to initiate irradiation degradation phenomena in the austenitic nickel-base alloys. The Combustion Engineering design surveillance tube holders are expected to accumulate the largest irradiation dose of the components considered, reaching approximately 3-5x10^{19} n/cm^2 after 60 calendar years of operation.

The report also discusses several options for addressing aging degradation, particularly for PWSCC. These options included primary water zinc additions, replacement of more susceptible materials with improved materials such as Alloy 690 or Alloy X-750 with a high temperature heat treatment, or other potential mitigation options.

EPRI Perspective

This report addresses the potential aging degradation of the nickel-based alloy components within the reactor vessels of Westinghouse and Combustion Engineering PWR designs. It is not intended to address all nickel-base alloy locations or all nickel-based alloys. Reactor vessel head penetrations, bottom-mounted instrumentation penetrations, and inlet and outlet nozzles have been excluded because PWSCC in these components has been addressed at great length in other studies. Components fabricated from Alloy 690 and its associated weld metals (Alloys 52/152) have also been excluded from this report because of their lower susceptibility to PWSCC, and because the MRP has ongoing projects that address them. EPRI recommends that utilities perform plant-specific design evaluations to determine which orphan locations apply and whether or not those components were fabricated from nickel-based alloys. Further evaluation should then determine what temperature those components experience during service in order to project a timeline for the onset of a potential concern for PWSCC initiation. Aside from these evaluations, the evidence presented in this report does not indicate a need for requirements beyond those contained in the NRC's *Generic Aging Lessons Learned* report. Maintenance planning would provide an effective mechanism for managing the effects of orphan location component material degradation due to PWSCC. Note that further research is required in certain areas, such as LTCP, to more fully understand the potential effects of aging on these components.

Keywords

Aging management, Aging degradation, Nickel-based alloys, Orphan locations,
Reactor vessel welded attachments, Irradiation effects, Primary water stress corrosion cracking,
Low temperature crack propagation, Wear

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1 INTRODUCTION

The aging degradation of austenitic nickel-based alloys and their associated weld materials, particularly through primary water stress corrosion cracking (PWSCC), represents a major cause of component degradation in pressurized water reactor (PWR) systems. Since 1986, PWRs, both in the U.S. and abroad, have experienced pressure boundary leaks as a result of PWSCC of Alloy 600 components or Alloy 82/182 weld metals. PWSCC has impacted the reliability of critical safety systems and required costly repairs and replacements of the affected components. Due to safety concerns, these events have prompted the U.S. Nuclear Regulatory Commission (NRC) to issue a number of documents, which have been complemented by documents prepared by the Materials Reliability Program (MRP) that is managed by the Electric Power Research Institute (EPRI). (See References 1–5.)

The current report addresses the potential for aging degradation of the reactor vessel welded attachments and a few other components inside the reactor vessel that are fabricated from austenitic nickel-based alloys and their associated weld metals. Figure 1-1 demonstrates the typical locations of nickel-based alloy materials in a Westinghouse-designed PWR primary loop. The typical locations of the nickel-based alloy materials in a Combustion Engineering (CE)-designed PWR primary loop are displayed in Figure 1-2.

The components examined in this report are sometimes called the nickel-based alloy orphan locations. These are locations in the reactor vessel, fabricated primarily from Alloy 600 and its associated weld metals, Alloy 182/82, which have not been addressed in previous evaluations. Components fabricated from Alloy 690 and its associated weld metals (Alloys 52/152) have been excluded from consideration because of their lower susceptibility to PWSCC, and because the MRP has ongoing programs that address them. Reactor vessel head penetrations, bottom-mounted instrumentation penetrations, and inlet and outlet nozzles have also been excluded because PWSCC in these components has been addressed at great length in other studies. The few components remaining after these filters have been applied are the orphan locations.

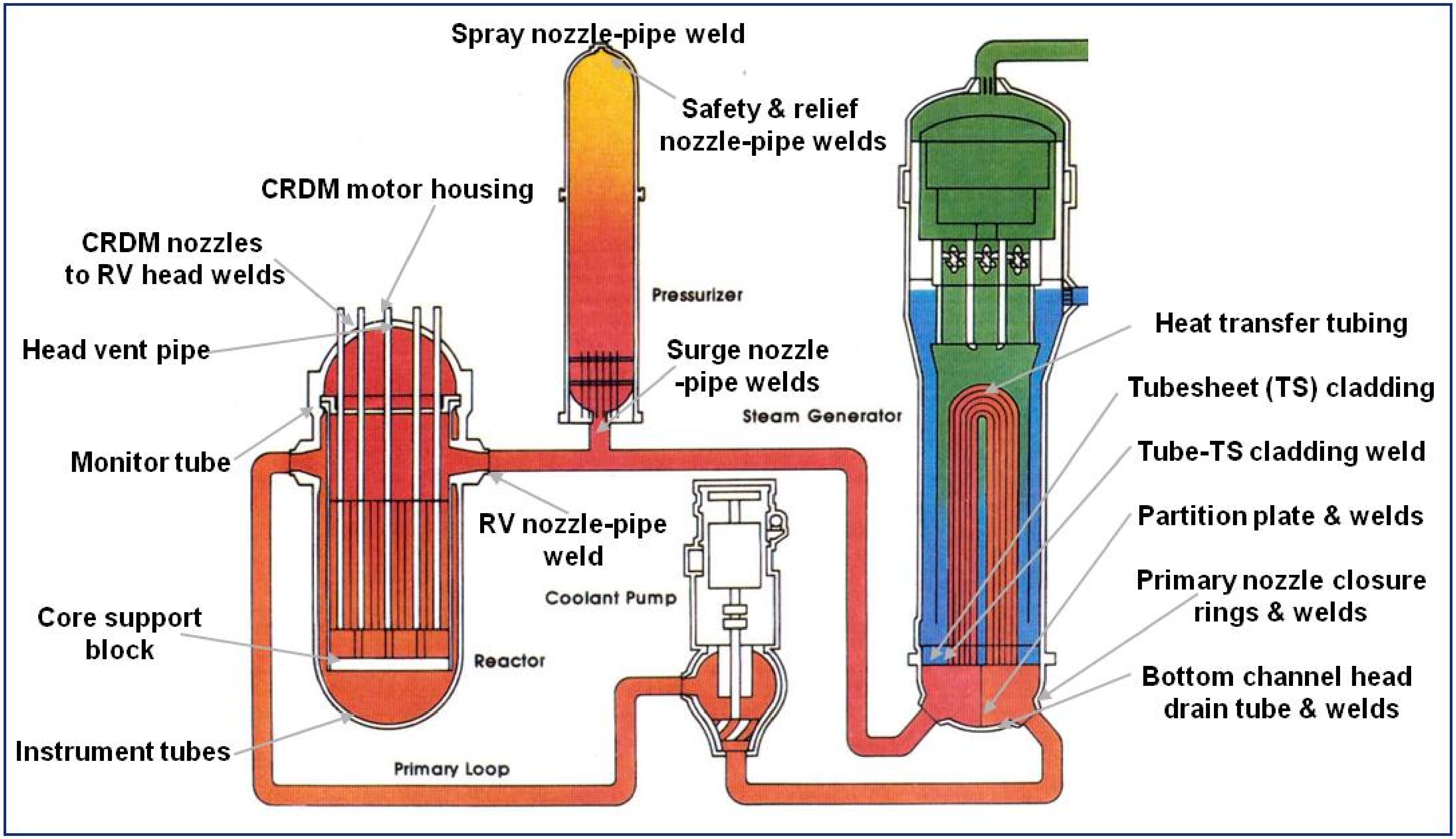

Figure 1-1
Primary Loop Locations Containing Nickel-Based Alloy Base or Weld Material in a Westinghouse-Designed PWR

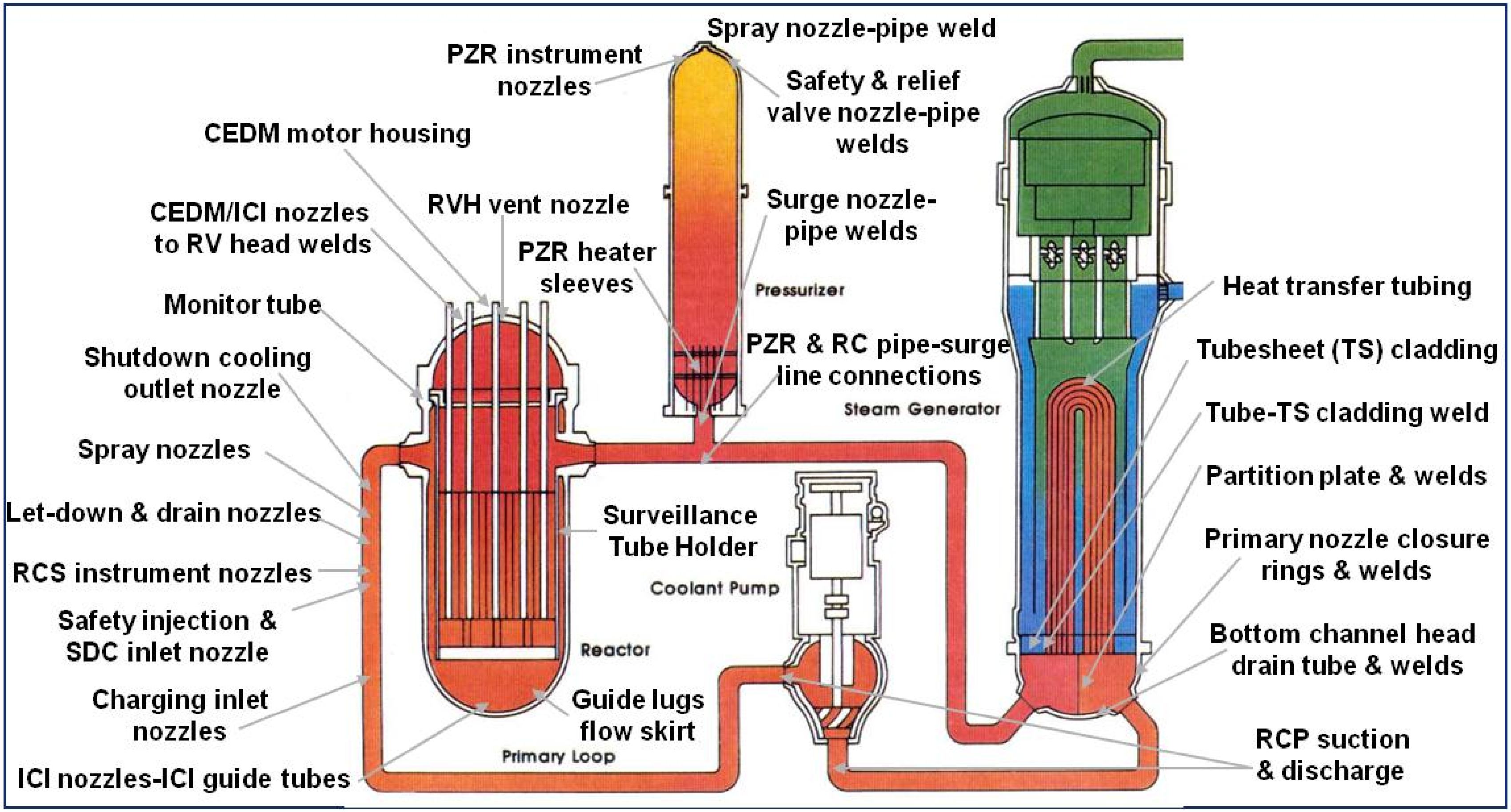

Figure 1-2
Primary Loop Locations Containing Nickel-Based Alloy Base or Weld Material in a CE-Designed PWR

2
THE ORPHAN LOCATIONS

2.1 Alloy 600 and Alloy 82/182 Welds

Table 2-1 and Table 2-2 list the orphan locations for the Westinghouse-designed and CE-designed PWRs, respectively. The tables give the material of fabrication and approximate temperature for each component.

The Alloy 600 type orphan locations in a Westinghouse-designed PWR are the core support lugs, the leak-off monitor tubes, and the clevis inserts. The core support lugs are typically welded directly to the reactor vessel by an Alloy 182/82 dissimilar metal weld and provide circumferential support and positioning for the core barrel through the clevis inserts. The clevis inserts are bolted to the core support lugs and interface with the radial support keys on the core barrel. The leak-off monitor tubes are positioned near the main reactor vessel O-ring seals. They are used to monitor for leak failures of the O-ring seals and are generally only wetted with primary coolant when a seal fails. The tubes are fabricated from Alloy 600 bar or tube material and are welded to the reactor vessel with an Alloy 182/82 weld.

The approximate temperatures of the orphan locations in Westinghouse-designed PWRs are presented in Table 2-1. Most of the locations are in the downcomer region of the reactor vessel and, as such, would have a low temperature approximately equal to the reactor vessel inlet temperature, T_{COLD}. These locations include the core support lugs and welds, and the clevis inserts. The leak-off monitor tubes are near the upper plenum region of the reactor vessel and are expected to be at a higher temperature. This is estimated to be near the mean head temperature; however, this is likely to be a conservatively high estimate, since the monitoring tubes are removed from direct exposure to the primary coolant.

The Alloy 600 type orphan locations in a CE-designed PWR are the core stabilizing lugs and stop lugs, the core stabilizing lug shims, the flow skirt, the leak-off monitor tubes, and the various parts of the surveillance tube holders. The core stabilizing lugs are part of the support and alignment system for the core barrel and consist of six pads fastened to the inside surface of the vessel wall by full-penetration Alloy 182/82 welds. The pads are equally spaced on the inside surface of the vessel shell near the juncture of the shell to the bottom head. Alloy 600 core stabilizing lug shims are bolted to the surfaces of the stabilizing lugs that interface with the core support barrel snubber lugs. This system allows vertical movement of the core barrel but prohibits movement in the circumferential direction and limits the amplitude of potential flow-induced vibration of the barrel. The core stop lugs are welded to the reactor vessel with Alloy 182/82 and limit the downward drop that the core could experience if the core support barrel were to fail. The flow skirt (also called the flow baffle) is welded to the reactor vessel bottom head and is a perforated cylinder designed to provide a uniform flow distribution to the reactor core and to minimize pressure drop. The surveillance tube holders are made up of several subcomponents that contain Alloy 600, including the surveillance tube holder, brackets, pins, and gussets. Six surveillance tube holders are welded to the reactor vessel in the core beltline region and hold the surveillance capsules that are used to monitor irradiation embrittlement of the reactor vessel materials. The seal leak-off monitor tubes serve the same function in the CE design as they do in the Westinghouse design – monitoring for leakage into the space between the two vessel O-ring seals. The tubes are fabricated from Alloy 600 material and welded to the reactor vessel with Alloy 182/82.

The approximate temperatures of the orphan locations in a CE-designed PWR are listed in Table 2-2. As with the Westinghouse design, most of the components are in the downcomer region of the reactor vessel. All of those components will have a temperature approximately equal to the inlet temperature of the reactor or T_{COLD}. The only exceptions to this are the leak-off monitor tubes, which are closer to the upper plenum of the reactor vessel and would have a conservative maximum temperature approximately equal to the mean head temperature.

Since they are within the reactor pressure vessel, the orphan locations are each subject to some level of neutron irradiation. Of all the components in both the CE and the Westinghouse designs, the surveillance tube holders are in closest proximity to the core and would receive the highest neutron fluence. After 60 years of plant operation, the CE surveillance tube holders are expected to have accumulated approximately $3\text{-}5 \times 10^{19}$ n/cm^2. Every other orphan location is expected to receive less than this value after 60 years of operation. To find the plant-specific fluence values for the surveillance tube holders, the end-of-life vessel internal diameter fluence should be consulted.

It should be noted that the component lists on the schematics Figure 1-1 and Figure 1-2, and in Table 2-1 and Table 2-2, are generic for the Westinghouse or CE fleets. Some of the components may not be present in every Westinghouse or CE reactor design, and some components may be fabricated from other materials (e.g., austenitic stainless steels) in certain plants. It is recommended that each individual power plant do a plant-specific confirmation of its particular configuration to determine materials of construction for the orphan locations.

2.2 Alloy X-750

Components fabricated from Alloy X-750 are not included in Table 2-1 and Table 2-2. Alloy X-750 is susceptible to some of the same aging degradation mechanisms as Alloy 600 and its weld metals. Alloy X-750 was primarily used in certain bolting applications inside of the reactor vessel.

In the Westinghouse design, Alloy X-750 bolts fasten the clevis inserts to the core support lugs, and Alloy X-750 was the material chosen for the control rod drive mechanism (CRDM) support pins. Cracking of the CRDM support pins has been an issue since the early 1980s. Replacement of the support pins with pins fabricated from superior materials has been the approach to dealing with this issue and has been implemented in many plants to address actual and potential aging degradation. Aging degradation of the clevis insert bolts was anticipated as a potential issue in the development of the reactor internals inspection and evaluation guidelines, MRP-227 [6]. The bolts were included as an Existing Component inspection in MRP-227, since the American Society of Mechanical Engineers (ASME) Code Section XI inservice inspection of the core support lug should detect any problems with the clevis insert bolts. Recent experience [7] has confirmed that there is a need to inspect this component for potential aging degradation and has also confirmed that the ASME Code Section XI inspection is adequate to detect degradation of these bolts.

CE-designed reactors vessels contain Alloy X-750 in a similar location to that of the clevis insert bolts in the Westinghouse design. These “core stabilizing shim-to-lug cap screws” were fabricated from Alloy X-750 and fasten the core stabilizing shims to the core stabilizing lugs. These bolts were not included as an Existing Component inspection in MRP-227. However, just as with the clevis insert bolts, aging degradation of the core stabilizing shim-to-lug cap screws would be detected by ASME Code Section XI inservice inspection of the core stabilizing lugs.

As knowledge of the stress corrosion cracking susceptibility of Alloy X-750 has increased, the heat treatments applied during fabrication have evolved. Earlier, lower temperature heat treatments such as the AH and BH treatments were eventually replaced by the higher temperature HTH heat treatment. Thus, particularly with the replacement guide tube support pins, there is significant variation in the heat treatments used for these Alloy X-750 components. Components fabricated earlier were likely fabricated from lower heat treatment material, while components fabricated later were probably fabricated from the HTH material. Plant records should be consulted to determine the actual heat treatment utilized for a particular component.

Table 2-1
List of Austenitic Nickel-Based Material Orphan Locations in Westinghouse-Designed PWRs

Component/Subcomponent	Material	Temperature
Core Support Lugs:		
Core support lugs	Alloy 600	T_{COLD}
Core support lug weld	Alloy 182/82	T_{COLD}
Clevis Inserts:		
Clevis inserts	Alloy 600	T_{COLD}
Leak-Off Monitor Tubes:		
Leak-off monitor tubes	Alloy 600	Mean head temp.
Leak-off monitor tube welds	Alloy 182/82	Mean head temp.

Table 2-2
List of Austenitic Nickel-Based Material Orphan Locations in CE-Designed PWRs

Component/Subcomponent	Material	Temperature
Core Stabilizing Lugs:		
Core stabilizing lugs	Alloy 600	T_{COLD}
Core stabilizing lug welds	Alloy 182/82	T_{COLD}
Core stabilizing lug shims:		
Core stabilizing lug shims	Alloy 600	T_{COLD}
Core Stop Lugs:		
Core stop lugs	Alloy 600	T_{COLD}
Core stop lug welds	Alloy 182/82	T_{COLD}
Flow Skirt:		
Flow skirt	Alloy 600	T_{COLD}
Flow skirt welds	Alloy 182/82	T_{COLD}
Leak-Off Monitor Tubes:		
Leak-off monitor tubes	Alloy 600	Mean head temp.
Leak-off monitor tube welds	Alloy 182/82	Mean head temp.
Surveillance Tube Holders:		
Surveillance tube holder	Alloy 600	T_{COLD}
Surveillance tube brackets	Alloy 600	T_{COLD}
Surveillance tube pins	Alloy 600	T_{COLD}
Surveillance tube gussets	Alloy 600	T_{COLD}

3
AGING DEGRADATION OF THE ORPHAN LOCATIONS

The orphan location components were considered for potential susceptibility to a number of aging degradation mechanisms. The irradiation effects of irradiation-assisted stress corrosion cracking (IASCC), irradiation-induced void swelling, irradiation-induced stress relaxation, and irradiation embrittlement were included as well as PWSCC, low-temperature crack propagation (LTCP) and wear.

3.1 Irradiation Effects

The potential irradiation effects on the orphan locations include IASCC, irradiation-induced void swelling, irradiation-induced stress relaxation, and irradiation embrittlement. Even when the CE-design surveillance tube holders are considered, the expected 60-year fluence of approximately 3-5x10^{19} n/cm^2 is not expected to be enough to cause IASCC or irradiation-induced void swelling. Additionally, the temperature at the surveillance tube holders is T_{COLD}, which also reduces the effects of these two mechanisms.

Irradiation-induced stress relaxation is not expected to be an issue for the orphan locations, because they are distant from the core and because most of the components are welded attachments. Even if some stress relaxation does occur, it would only be expected to reduce potential residual stresses in the welds. Relaxation in the components would not result in a loss of preload leading to wear or fatigue.

Irradiation embrittlement may occur in the most highly irradiated orphan location component, the surveillance tube holders. However, this would not require additional inspection. If cracking were discovered in the surveillance tube holders, any potential irradiation embrittlement would have to be considered when calculating the flaw tolerance of the holders.

3.2 Low Temperature Crack Propagation

LTCP is a phenomenon that occurs in some nickel-based alloys in the presence of hydrogen environments between the temperatures of approximately 50°C and 150°C. This phenomenon was originally observed in nickel-based alloys, such as Alloy 82, by Mills and Brown at Bettis [8, 9, 10]. Since those initial results, other work has been done to determine which alloys may be affected by LTCP. The published laboratory results have shown that LTCP is not an issue for the base metals Alloy 600 and 690, but it can occur in other nickel-based alloys such as Alloy X-750 and the weld materials Alloy 182/82 and Alloy 152/52 [11, 12, 13].

Under normal operating conditions, components in the reactor pressure vessel of a PWR are not subject to the temperatures at which LTCP occurs. However, these temperatures can occur during shutdown and, depending on the plant startup and shutdown procedures, may occur along with elevated levels of environmental hydrogen. One important aspect of LTCP is that it is a mechanism for crack growth but not for crack initiation. LTCP will not occur unless some other mechanism, such as stress corrosion cracking (SCC) or fatigue initiates a crack.

LTCP is mentioned here because of the developing database indicating that it may be an issue for nickel-based PWR components. However, it will not be given further consideration in this report, because it is not yet understood well enough to draw conclusions or develop guidance. Research is ongoing to develop the relevant experimental database necessary to better understand the LTCP phenomenon.

3.3 Primary Water Stress Corrosion Cracking

Essentially all nickel-based materials (Alloy 600, Alloy X-750, Alloy 718, Alloy 182, Alloy 82, et al.) that experience service in high-temperature primary water under high applied-plus-residual stress must be considered as susceptible to PWSCC based on operating experience. The apparent exceptions to this generalization are Alloy 690 and its associated weld metals Alloys 52 and 152. This exception appears to result from the exceptionally high chromium content (27wt% to 31wt%) in these alloys. Components fabricated from Alloy 690 have been exempt from the current evaluation for this reason. Previously established laboratory test data have clearly shown that PWSCC of austenitic nickel-based alloys is a thermally activated process and that the initiation rate can be described by the expression:

$$\text{Initiation Rate} = \frac{1}{t} = A\sigma^{n} e^{\frac{-Q}{RT}} \qquad \textbf{Equation 1}$$

where:

A is a term that describes the microstructural and other material conditions of the material

σ is the effective stress term (resulting from applied and residual stresses) in MPa

n is the stress exponent; a value of 4 is used in this assessment based on laboratory test data

Q is the activation energy for the crack initiation process; a value of 50 kcal/mole is used for this assessment (this same approach has been used in EPRI MRP reports on Alloy 600 [14, 15])

R is the gas constant (1.987 cal/mole-K)

T is the absolute temperature in K (°C + 273.15)

t is the time to initiate cracking

SCC only occurs when tensile stress, a corrosive environment, and a susceptible material are combined. If any one of these three conditions is not met, SCC will not occur. Experience has shown that significant and simultaneous contribution by each of the material, stress, and temperature factors in Equation 1 is essential for SCC, specifically PWSCC when the environment is PWR primary water, to occur.

In general, a microstructure with continuous grain boundary carbides is considered to have a greater resistance to PWSCC, while a microstructure with a lower density of intergranular carbides is less resistant to PWSCC. This has been consistently borne out in laboratory tests and in examinations of pulled steam generator tubes. Since PWSCC follows an intergranular morphology, a larger grain size is favored for better resistance. A larger grain size corresponds to a smaller grain boundary area, which favors better grain boundary coverage of carbides for a given carbon content. A larger grain size would also provide better PWSCC resistance because of lower yield strength. A fine grain size results in higher yield strength, which may result in higher residual stresses and reduced PWSCC resistance.

To a lesser-defined extent, the general morphology of the material – i.e., the orientation of the microstructure with respect to the effective stress – may also play a role. This may be a contributing factor to the generally greater resistance to crack initiation that is observed in welds in which the cracks must initiate and propagate along interdendritic boundaries.

Equation 1 indicates that the time to crack initiation varies inversely with the fourth power of stress. Thus, a 50% reduction in the effective stress increases the time to initiate cracking by a factor of 16. MRP-175 [16] gives screening criteria for the potential onset of PWSCC. According to these criteria, Alloy 600 base metal is susceptible to PWSCC at effective stresses beyond 30 ksi, and its weld materials are susceptible above 35 ksi. Stresses below these conservative screening criteria are not expected to cause PWSCC.

The net effective stress includes contributions due to residual, operating, and/or thermal stresses. Component fabrication processes such as welding, rolling, reaming, bending, and other forms of cold work will introduce residual stresses in the material that may contribute to PWSCC. Other conditions remaining constant, higher yield strength material is likely to retain higher residual stresses from fabrication and, hence, may crack in a shorter period of time. In many cases, local (residual) stresses introduced by cold work and welding during fabrication are more important than service stresses, since service stresses are generally well below the yield stress.

Laboratory data show that because of temperature differences between various components or locations, the time to initiate PWSCC can vary significantly due to environment. For example, for temperatures in the vicinity of 610°F (321°C), application of the generally accepted activation energy of 50 kcal/mole indicates cracking kinetics will vary by a factor of two for a temperature change of 10°C (18°F). The majority of cracking has occurred in locations subject to higher system operating temperatures. There have been few instances of PWSCC at temperatures typical of T_{COLD} conditions. Table 3-1 demonstrates the effect of varying the temperature on the crack initiation rate relative to a temperature of 600°F (315.6°C).

3.4 Wear

Wear is another mechanism that should be considered when addressing the potential aging degradation of the orphan location components. However, it is not expected to be an issue for the components in either the CE or Westinghouse design except for the cases of the clevis inserts in the Westinghouse design and the core stabilizing lug shims in the CE design. The wear could either result from interaction between the insert or shim and the mating part on the core barrel or from movement of bolts after a loss of preload. The rest of the orphan locations are typically not in the vicinity of parts subject to relative motion either during operation or refueling. The clevis inserts and core stabilizing lug shims are inspected under the ASME Code Section XI inservice inspections, which will determine if there is unacceptable wear of these components.

Table 3-1
PWSCC Initiation Kinetics for Alloy 600 as a Function of Temperature

Temperature		Relative SCC Initiation Rate
°F	°C	
500	260.0	0.012
510	265.6	0.019
520	271.1	0.030
530	276.7	0.048
540	282.2	0.077
550	287.8	0.120
560	293.3	0.186
570	298.9	0.287
580	304.4	0.438
590	310.0	0.663
600	315.6	1.000
610	321.1	1.486
620	326.7	2.200
630	332.2	3.234
640	337.8	4.719
650	343.3	6.840

4
PATHS TO DISPOSITION PWSCC CONCERNS

Mitigation of cracking or potential cracking in the orphan locations will be required if material degradation as a result of PWSCC would result in safety or reliability concerns. As noted in Section 2, PWSCC can only occur when tensile stress, a corrosive environment, and a susceptible material are all present. Alloy 600 and its weld metals, Alloy 182/82, are susceptible to PWSCC and meet the material requirement, so the question of whether or not there is concern due to PWSCC in the orphan location components must be approached through evaluation of the magnitude of the tensile stresses present and the details of the environmental conditions (specifically the temperature of the component).

Fabrication residual stresses in the components listed in Table 2-1 and Table 2-2 are expected to be low for the base metal remote from the welds and higher for the welds and the base metal near the welds. Components that have been annealed prior to installation would have minimal residual stress. Welds are generally assumed to produce residual stresses in the weld metal and adjacent base metal approximately equivalent to the yield strength of the material. The welds listed in Table 2-1 and Table 2-2 are between the component and the reactor vessel. The weld-induced residual stresses in the Alloy 600 would have been reduced by the post-weld stress relief heat treatment (PWHT) of the reactor vessel during fabrication. It is typically assumed that PWHT reduces the weld residual stresses by approximately half of the material yield strength. In combination with the service stresses of these components, the orphan locations will experience tensile stresses that are not expected to drive PWSCC. However, the environment, particularly the low temperature, is the key factor making PWSCC unlikely.

PWR primary water is a corrosive environment capable of causing PWSCC in Alloy 600-type materials. All of the components except for the leak-off monitor tubes are located in the coolant downcomer region of the reactor vessel. The primary water in this region is at the reactor vessel inlet or cold leg temperature, which is 560°F (293°C) or less for most PWRs. It should be noted that for all of the cases except the leak-off monitor tubes, the orphan locations are exposed to flowing primary water and not stagnant conditions where impurity ions or oxygen could concentrate. The temperature in Equation 1 has a strong effect on the time to PWSCC initiation as shown in Table 3-1. An orphan location at 560°F (293°C) is expected to initiate PWSCC a factor of 5.3 times slower than a similar component at a temperature of 600°F (315.6°C). The time to PWSCC initiation calculated for 560°F (293°C) would be conservative for a plant operating at a lower inlet temperature.

MRP-48 [17] discusses the "Effective Degradation Year" (EDY) concept as a way to compare the relative time-temperature "age" of components susceptible to PWSCC. This concept uses relative initiation rates such as those given in Table 3-1 to compare the degradation of a specific component to that of a component at the reference temperature of 600°F (315.6°C). For example, a component that has experienced 10 effective full-power years (EFPY) at 600°F (315.6°C) has experienced 10 EDY, while a component that has experienced 10 EFPY at 560°F (293°C) has experienced 1.86 EDY:

$$1.86 \ \mathrm{EDY} = \left[10 \ \mathrm{EFPY}\right] \times \left[0.186 \frac{\mathrm{EDY}}{\mathrm{EFPY}}\right] \quad \textbf{Equation 2}$$

This is useful in determining when a susceptible component may begin to show evidence of PWSCC. Operating experience data indicate that PWSCC in Alloy 600 and its weld materials generally initiates no earlier than after 8–10 EDY. For even the most conservative case of expecting initiation at 8 EDY for a component at 560°F (293°C), the earliest PWSCC would not be expected until after 43 EFPY:

$$43 \ EFPY = \frac{8 \ EDY}{0.186 \dfrac{EDY}{EFPY}} \quad \textbf{Equation 3}$$

For those plants where the inlet temperature is lower, crack initiation is not expected until after a greater number of EFPY. Table 4-1 gives examples of when 8 EDY would be reached in a component across a range of temperatures. From a temperature standpoint, the potential for crack initiation would not be expected until well into the license extension period.

The leak-off monitor tubes have been mentioned in this section as an exception to the general conditions of low temperature and flowing coolant. They are conservatively expected to have a maximum temperature of approximately the mean head temperature due to their location. However, they are not expected to experience PWSCC because under normal conditions they are not in direct contact with primary water. The monitor tubes are located between the main sealing O-rings of the reactor vessel and are exposed to primary coolant only when a seal fails and leaks.

One issue that could be a concern for the orphan location components welded to the reactor vessel is the potential for a crack that would initiate and grow through the component by PWSCC and then continue into the reactor vessel material by fatigue crack growth. If this occurred, it could lead to a failure of the reactor vessel. Based on the evaluations already given in this section, a crack growing into the reactor vessel material is unlikely. However, if such a crack were present, it would not be expected to cause a failure of the reactor vessel based on the results of the PWR Owners Group program to investigate risk-informed extensions of reactor vessel inservice inspection intervals (WCAP-16168-NP-A [18]). The results of this program showed that significant fatigue crack growth did not occur between inservice inspections. The program found that the applied stress intensity of a hypothetical fatigue crack in the pressure vessel did not vary significantly with time. Note that WCAP-16168-NP-A determined that the limiting location in the reactor vessel was the beltline region. Of the orphan locations, only the surveillance tubes are in the beltline region. Thus, the results of the risk-informed inservice inspections program would be conservative relative to all but the surveillance tube orphan locations.

The "Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure," MRP-156 [19], describes a number of orphan location components and gives the expected consequence of failure. The Issue Management Table (IMT) [20] uses generic terminology for many components, since similar components exist across the various PWR designs yet often have different names. For example, the table lists the core guide lugs for the reactor vessel. This component covers the core support lugs for Westinghouse-designed reactors and the core stabilizing lugs and core stop lugs for the CE-designed reactors. MRP-156 lists the core guide lugs, CE flow baffle (flow skirt), specimen surveillance holder tubes, and leakage monitoring tubes as having no consequence of failure. The consequences of failure considered in the formation of the IMT ranged from safety consequences to economic consequences. Of the orphan locations, only the clevis inserts were highlighted as having a consequence of failure. MRP-156 stated that the replacement of the clevis inserts could result in a significant economic impact for the plant, though it would not represent a safety hazard if an insert failed during operation. A similar economic impact would be expected for the replacement of the core stabilizing lug shims in the CE design.

In summary, concern due to PWSCC for the orphan locations will not require actions on any level until approximately 43 EFPY under normal operational conditions. After this point, there may be potential for PWSCC initiation depending on the temperature conditions of the plant and the specific stress levels of the component. The orphan locations have such a long initiation time because of the low service temperatures that they experience. For the case of the leak-off monitor tubes, the lack of exposure to primary coolant will prevent the initiation of PWSCC. Even if a crack were to initiate in any of the orphan locations prior to or after 43 EFPY, the low temperatures would result in low crack growth rates. Based on the results of the PWR Owners Group program on the risk-informed assessment of inservice inspection interval extensions, a crack in an orphan location that reaches the reactor vessel is not expected to cause a concern for reactor vessel pressure boundary failure. It is also important to note that the IMT has addressed the consequences of failure for the orphan location components and states that there are no expected safety consequences for failure of any orphan location component and an economic consequence for failure of the clevis inserts only, which is also applicable to the core stabilizing lug shims. Based on these conclusions, mitigation of potential PWSCC could be effectively managed through continued inspection (where applicable) and maintenance planning.

Table 4-1
EFPY Required to Reach 8 EDY

Temperature		EFPY to 8 EDY
°F	°C	
530	276.7	165.00
535	279.4	131.09
540	282.2	104.39
545	285.0	83.32
550	287.8	66.65
555	290.6	53.43
560	293.3	42.92
565	296.1	34.56

5 POTENTIAL ACTIONS TO ADDRESS PWSCC CONCERN

PWSCC of nickel-based austenitic alloys has caused a significant number of problems for the industry over the past several decades. Most of these problems have been in steam generators, pressurizers, and dissimilar metal welds; however, the various incidents have raised concerns for other locations such as the orphan locations. Many research and evaluation programs have been conducted to determine strategies for managing PWSCC in affected components. A selection of these mitigation techniques are described here. The MRP has created several documents [21, 22] to summarize these and other mitigation techniques as well as to give guidance for the management of Alloy 600 components.

5.1 Zinc Chemistry

Laboratory and PWR field data have shown that the addition of zinc to primary water inhibits PWSCC crack initiation [23, 24]. The exact factor of improvement due to zinc additions has yet to be determined; however, reductions by factors of 2 to 10 have been observed in the results from steam generator tube field inspections. The results of extensive laboratory testing are consistent with these field data. There are also some data showing that zinc may be effective in reducing crack growth rates, but the results from crack growth rate experiments are mixed. Aside from the benefits to PWSCC initiation and the potential benefits to crack growth, zinc has proven to be beneficial in reducing the impact of radiation [23, 24].

A number of U.S. and international utilities have implemented zinc additions as part of their primary coolant chemistry. Some are adding the zinc primarily for the reduced radiation impact, while others are adding zinc with the intention of reducing PWSCC initiation. EDF, for example, has begun implementing zinc at several plants and intends to implement it throughout its fleet with purposes varying depending on the experience at the particular plant [25]. Farley Unit 2 saw an apparent benefit in the PWSCC initiation for its Alloy 600 CRDM head penetrations as a result of zinc addition. The heat of Alloy 600 material used for the Farley Unit 2 CRDM penetrations was also used at several other PWRs. In each of those other PWRs that did not add zinc, the material experienced a significant level of PWSCC. The CRDM penetrations at Farley Unit 2 did not experience PWSCC, which suggests that zinc had a beneficial effect in preventing PWSCC at Farley [23].

It is also anticipated that Alloy 690 materials will receive the same improvement in PWSCC initiation as Alloy 600 materials if problems with PWSCC are observed in Alloy 690. However, more research to support this assumption is still required [23].

The apparent benefits for PWSCC and radiation release due to zinc occur as a result of the way it interacts with the oxide films of wetted surfaces in a PWR primary loop. The zinc is incorporated in the oxide films and displaces nickel and cobalt from their crystalline lattice sites. This occurs primarily in the inner layer of oxide and makes the oxide more resistant to continued corrosion. The replacement of the cobalt on the lattice sites reduces the radioactivity contained in the oxide and the more protective oxide layer reduces PWSCC initiation.

Due to the beneficial effects of zinc, a number of utilities are already adding zinc to the primary coolant. EPRI has issued the “Pressurized Water Reactor Primary Water Zinc Application Guidelines” [23] to guide the implementation of zinc chemistry. EPRI’s “Pressurized Water Reactor Primary Water Chemistry Guidelines,” [26] identifies zinc addition as a “Recommended Element,” consistent with NEI 03-08 [28], of a Strategic Water Chemistry Plan because of the benefits for dose rate reduction and PWSCC mitigation.

5.2 Mitigation, Repair, or Replacement with Improved Materials

One approach to addressing the aging degradation of nickel-based alloys is through the mitigation, repair, or replacement of a component with improved materials. For Alloy 600, the material of choice is typically Alloy 690 and its weld materials. For Alloy X-750, bolting materials have been replaced with stainless steel such as cold worked Type 316 or with improved heat treatment Alloy X-750 material.

Alloy 690 is an austenitic nickel-based alloy with a proven resistance to PWSCC in PWR primary water conditions. This is attributed to the high chromium content (27wt%–31wt%) of the alloy and of its weld metals, Alloys 52 and 152. A significant number of plants have replaced their steam generators or reactor vessel closure heads with new components substituting Alloy 690 for the parts originally fabricated from Alloy 600.

Alloy X-750 is a relatively mature alloy and has gone through a number of different heat treatment iterations. The earliest heat treatments were at lower temperatures and were particularly susceptible to PWSCC. Later heat treatments included a higher temperature annealing step. These HTH heat treatment Alloy X-750 materials have improved stress corrosion cracking resistance as compared to the earlier versions [27]. Replacement of Alloy X-750 by stainless steel components has been implemented in certain cases. However, the difference in strength between Alloy X-750 and stainless steel precludes the use of stainless in some cases. Since Alloy X-750 is used in bolting applications, replacement will be the primary method for addressing actual degradation or potential PWSCC concerns.

MRP-258 [29] summarizes much of the knowledge about the resistance of Alloy 690 to PWSCC. It concludes that, under most conditions in PWR primary water, Alloy 690 provides a conservative factor of improvement of 40 to 100 times more resistance to PWSCC initiation than mill-annealed Alloy 600. For crack growth rates, factors of improvement for Alloy 690 relative to mill-annealed Alloy 600 could be greater than 100, although the extremely low crack growth rates in Alloy 690 have made precise determination of this value difficult.

MRP-237 [30] includes some evaluation of the weld materials Alloys 52 and 152. The report concludes that these materials are also expected to have high resistance to PWSCC when exposed to PWR primary water. They provide a minimum factor of improvement for PWSCC initiation of at least 60 times relative to Alloy 600 materials. The resistance of Alloy 52/152 to crack growth is more complex and depends to some extent on the quality of the weld and the effect of the dilution zone near the fusion line of dissimilar metal welds. Factors of improvement as high as 400, relative to Alloy 182, have been measured; however, due to some test results that gave higher crack growth rates the minimum factor improvement is 70 per MRP-237.

Components that could be easily replaced by new components fabricated from Alloy 690 would benefit from the significantly higher PWSCC resistance of the alloy, relative to Alloy 600. However, for many of the orphan location components, replacement is probably not a feasible option due to location or the means by which the component is attached to the reactor vessel. If a problem were discovered, it is more likely that it would be corrected by weld repair for most of the orphan locations. Several different methods of weld repair are described in MRP-118 [22] for Alloy 600. Methods such as laser cladding and gas tungsten arc welding could be used to repair flaws and provide a more PWSCC and corrosion-resistant layer of Alloy 52/152 weld material. The protective cladding of weld material isolates the susceptible Alloy 600 or Alloy 182/82 base from the corrosive environment, reducing or preventing further PWSCC degradation. MRP-118 lists results from testing of Alloy 600 materials after cladding with resistant weld material. Generally, the results for the clad material showed improvement over the base metal; though, the report does state in several of the cases that the test results are probably too conservative due to the testing geometries employed. Weld cladding with more resistant material could also be applied proactively to protect Alloy 600 components and welds.

In summary, the use of Alloy 690 materials for replacements, repairs, or mitigation could provide an effective means of dealing with PWSCC of Alloy 600 orphan locations. The use of improved heat treatment Alloy X-750 can provide similar benefits for the relevant bolting applications. Though there are some locations that could be addressed by replacement with Alloy 690, it seems likely that the preferred method would be weld repair and cladding. Since Alloy X-750 is used in bolting applications, the primary method of dealing with degradation would be replacement of the bolts. Proactive cladding of the components with Alloy 690 material or replacement of Alloy X-750 bolting materials could be employed if there is concern that a component will experience PWSCC.

5.3 Other Mitigation Options

There are a number of other options available to address PWSCC of the Alloy 600 orphan locations. Many of these are presented in detail in MRP-118 [22]. These include options for surface conditioning and for protecting the surface by spraying or welding. Technologies such as cavitation peening, low plasticity burnishing, laser peening, nickel plating, and flapper wheel grinding are discussed. Where test data were available at the time of the report issuance, they were also included.

Other options that have been developed or are under development for mitigation of PWSCC in Alloy 600 components include laser peening and nickel plating of the component with a cold spray process. MRP has been investigating surface treatments including laser peening and has published a technical basis document for primary stress corrosion cracking mitigation by surface treatments [33]. The nickel plating protects the Alloy 600 component by isolating it from the corrosive environment behind a corrosion-resistant layer. Experience with this process for the nuclear industry is limited, but there is experience with using cold spray processes to deposit corrosion-resistant coatings for high temperature applications, particularly in aircraft turbines [31]. Cold spray is an attractive option because it is cost-effective and relatively easy to apply. Additional research on nuclear-specific applications would solidify supporting technical evidence and enhance the value of cold spray techniques for Alloy 600 PWSCC mitigation.

6 CONCLUSIONS

The Alloy 600 orphan locations typically include three components for the Westinghouse-designed reactor and six components for the CE-designed reactor. For Westinghouse, these are the leak-off monitor tubes, core support pads, and clevis inserts; and for CE, they are the core stop lugs, core stabilizer lugs, core stabilizer lug shims, leak-off monitor tubes, flow skirt, and surveillance tube holders. The potential for aging degradation due to irradiation effects, PWSCC, LTCP, and wear were considered at each of these locations and only PWSCC was determined to be a potential concern.

PWSCC occurs when a susceptible material, corrosive environment, and tensile stress are present concurrently. For the orphan location components, PWSCC is a potential concern due to the susceptibility of Alloy 600 type materials in primary water conditions, especially at locations where weld residual stresses are present to increase the overall effective applied stress. However, the temperature of all of the orphan location components, except the leak-off monitor tubes, is expected to be low, which will delay the potential for PWSCC crack initiation and slow crack growth. For the components with temperatures at T_{COLD}, PWSCC initiation should not be a concern before 43 EFPY. This will occur later for plants with inlet temperatures lower than the maximum of 560°F (293°C) assumed in calculating 43 EFPY. If a crack does occur, the low temperature would also result in a low crack growth rate. For the leak-off monitor tubes, which are expected to have a maximum temperature below the mean head temperature, PWSCC is not a concern because they are not exposed to primary water under normal conditions. The monitor tubes will only be exposed to a corrosive environment when the main vessel O-ring seal leaks.

Orphan location components that are welded to the reactor vessel are not expected to pose a risk for reactor vessel failure based on the results of the PWR Owners Group program documented in WCAP-16168-NP-A [18]. Even if a crack were to propagate through the component or weld and reach the vessel, the results documented in WCAP-16168-NP-A indicate that significant fatigue crack growth into the pressure vessel steel will not occur and will not require corrective response.

A plant can mitigate cracking by employing a number of different options. Some of these are proactive methods for reducing or preventing PWSCC, while others are actions that can be taken once a crack is detected. Zinc additions and cladding are examples of mitigation techniques that were described in some detail in this report.

It is recommended that utilities perform plant-specific design evaluations to determine which orphan locations apply and whether or not those components were fabricated from nickel-based alloys. Further evaluation should then determine what temperature those components will see during service in order to project a timeline for the onset of a potential concern for PWSCC initiation. Aside from these evaluations, the evidence presented in this report does not indicate a need for requirements beyond those contained in the Generic Aging Lessons Learned report [32]. Maintenance planning would provide an effective mechanism for managing the effects of orphan-location component material degradation due to PWSCC.

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