NOV 22 1985

MEMORANDUM FOR: James P. Knight, Acting Director
Division of Engineering

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: SCHEDULE FOR RESOLVING AND COMPLETING GENERIC ISSUE 111, "STRESS CORROSION CRACKING OF PRESSURE BOUNDARY FERRITIC STEELS IN SELECTED ENVIRONMENTS"

The MTEB request for research on this issue is classified as a licensing issue (i.e., programmatic and not a significant public safety issue). The research results are needed to increase knowledge, certainty, and understanding of the potential for stress corrosion cracking of the ferritic pressure vessel steels in selected environments. Licensing issues are normally assigned NRR resources at the discretion of the assigned Division Director based on the limitations within the division after assigning resources to approved HIGH and MEDIUM priority generic safety issues.

While you agree that this issue is of minimal risk, you initially requested that RES implement and complete new research on this issue promptly. However, the current ASME code requirements appear to be adequate to limit the public risk. The economic penalty that results from shutting down and repairing cracks in steam generators should provide sufficient incentive for the industry to correct this problem. Industry has been informed through IE Information Notice 85-65.

Based on the above considerations, I am concurring on your revised research request to include this issue in the current RES research program on "Environmentally-Assisted Crack Growth In LWR Materials" with the understanding that this inclusion will result in no impact on cost or schedule. You should keep informed of the status of this issue.

The attached prioritization evaluation (Enclosure 1) will be incorporated into NUREG-0933, "Prioritization of Generic Safety Issues," and is being sent to the regions and other offices, the ACRS, and the PDR for comments

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Pipe Crack

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on the technical accuracy and completeness of the prioritization evaluation. Any changes as a result of comments will be coordinated with you.

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Prioritization Evaluation

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ISSUE 111: STRESS CORROSION CRACKING OF PRESSURE BOUNDARY FERRITIC STEELS IN SELECTED ENVIRONMENTS

Background

Indications of possible stress corrosion cracking in the Indian Point Unit 3 steam generator prompted MTEB/NRR to review foreign and domestic operating experiences related to possible indications of stress corrosion cracking (SCC) in low-alloy ferritic steels. The incidents identified in Reference 1 as possible precursors to generic concerns of SCC relate to BWR reactor vessels, and PWR steam generators. These events, and some additional information, that are reviewed and discussed in this evaluation include:

- A through-wall crack in the transition cone of the steam generator shell at Indian Point Unit No 3.
- A through-wall crack in the lower head closure weld region of the Italian Garigliano steam generator (an indirect cycle BWR similar to a BWR-1).
- A guillotine rupture of a transition cone (reducer) in the secondary piping of the German HDR test facility.
- Cracking of feedwater lines in Westinghouse PWRs.
- Other events that may contribute to SCC in the BWR reactor vessels and the PWR steam generator vessels.
- Inferences from materials testing.

The materials of interest are those low-alloy ferritic materials used in fabrication of the subject pressure vessels (SA-533-Grade B, SA-508-Grade 2, and SA-302-Grade B).

Safety Significance

The reactor vessels and steam generators are constructed of low-alloy ferritic steels, and designed to the ASME Codes. The ASME Codes are linked to fatigue crack initiation in chemically unreactive environments (ASME Section III), and fatigue crack growths of existing defects as part of the ASME Section XI inspection code. Even though a corrosion allowance is specified in the ASME Codes, as a design consideration, it is not linked to corrosion fatigue, or stress corrosion cracking, that may occur in active chemical environments such as those experienced in the nuclear pressure vessels (reactor pressure vessels, steam generator pressure vessels).

Should the materials used in the pressure vessels be susceptible to SCC and exceed the inherent allowances in the ASME design/inspection Codes, a vessel rupture could result in a core-melt and radiation doses to the public. This

issue affects the design and operation of all light water reactors, except the B&W plants (Ref. 34).

Possible Solution

Prior to developing a solution to this problem, MTEB proposed a research scoping effort to define the severity of the problem, and the conditions under which the SCC phenomena are likely to be exacerbated. The research effort would also involve laboratory testing of the low-alloy materials in reactor-grade water with variable oxygen, chloride, and copper as possible water chemistry constituents.

No risk reduction can be attributed to study (scoping) efforts. However, the proposed effort is expected to better define under what conditions SCC of the pressure boundary steels may occur, and if such conditions arise, or prevail, during reactor operations. The proposed effort would also involve determinations of the effectiveness of post-weld heat treatments and water chemistry excursions that may affect the materials resistance to SCC. The results of these studies (research) could then possibly be used to determine when and where to conduct inspections to detect the cracks before they become a safety concern.

Priority Determination

Background Frequency Estimates: This section reviews and discusses the incidents identified in the background to help develop background frequency information to establish the safety significance of this issue.

Indian Point Unit No. 3 Steam Generator Event

On March 27, 1982, during a refueling outage (with the reactor in a cold shutdown condition), a small leak was detected on the shell side of steam generator #32 of the Indian Point 3 (IP-3) plant. The leak originated in the circumferential weld joining the transition cone to the upper shell. The steam generator shell is constructed of SA-302 Grade B material of 4" approximate thickness. To characterize the cracking phenomenon, the utility had various samples removed for metallurgical evaluation and failure analyses. Brookhaven National Laboratory (BNL) performed an independent failure analysis on specimens from steam generator #32 and on three additional boat samples containing cracks cut from steam generator #31. IE issued Information Notice 82-37 (Reference 5) to inform the industry of the event. In Reference 6, Westinghouse informed the NRC staff that no indications, or indications similar to those observed at Indian Point 3, were identified in the inspections performed on steam generators in 12 plants.

An investigation by BNL as reported in NUREG/CR-3281 (Reference 7) concluded that the cracking was caused by a low cycle corrosion fatigue phenomenon with

cracks initiating at areas of localized corrosion (pits) and propagating by fatigue. The cause of the pitting/cracking was considered to be related to the unit's relatively high operating dissolved oxygen levels and copper species in solution. The report also concluded that stress corrosion cracking could not be entirely discounted as possible failure mechanism. Reference 7 also identified that Indian Point-3 had developed moderate to severe denting of the SG tubes. The sludge analysis in Indian Point-3 showed concentrations of copper as high as 45% and iron 40%. Significant amounts of chlorine (Cl), copper as cuprous oxide (Cu_2O), and alpha hematite (alpha-Fe_2O_3) were also present in the sludge pile. The presence of these constituents indicated that water chemistry control in the Indian Point-3 steam generators had been poor for a considerable period of time. Additionally, in January 1981, IP-3 experienced a turbine blade failure which damaged approximately 50 condenser tubes and allowed chloride into the steam generators with recorded levels of up to 325 parts per million (ppm). The chloride intrusion may have had some influence in initiating pits at the inside surface of the steam generator shell.

Results from constant extension rate tests (CERT) on SA-302 Grade material in neutral and chloride solutions were reported in NUREG/CR-3614 (Reference 8). The CERT were performed on weld and base metal samples in air, water, and chlorine solutions. The chlorine solutions as Sodium Chloride (NaCl) and Cupric Chloride ($CuCl_2$) ranged from 1 ppm to 325 ppm chlorine. The results of the test indicated no significant effect in the NaCl CERT. However, the $CuCl_2$ CERT indicated possible susceptibility of the SA-302 Grade B material with as little as 1 ppm chlorine (as $CuCl_2$) in 268°C water. No attempt was made to control the dissolved oxygen content in the water. The combined results appear to indicate that copper as $CuCl_2$ may significantly alter the electrochemical reaction. The IP-3 secondary water chemistry may, however, provide an even different corrosion mechanism than that of the CERT. In this regard, the electromotive force series of metals could also produce galvanic corrosion of the iron (Fe) in the presence of copper (Reference 9) because carbon steel is anodic compared to copper (Cu) in the galvanic series. Thus, pitting/crevice corrosion of the carbon steel may have been acting as a combination of galvanic corrosion and low cycle fatigue. In the latter case, corrosion products in cracks (crevices) may act as wedges during cooldowns causing crack extensions. During heatups newly exposed crack surfaces develop more corrosion deposits. Repeated cycles, therefore, may result in through-wall cracks (corrosion-fatigue).

Because of the poor secondary water chemistry control at Indian Point 3, the atypical massive chloride intrusion, the results of the Westinghouse inspections on other steam generators, the event at IP-3 may not represent a generic PWR condition, but a plant-specific combination of atypical events. However, because of uncertainties in the CERT to represent conditions that may have prevailed at IP-3, and the indications from the CERT of the potential for copper in solution to effect some form of corrosion-related

attack on the low-alloy materials, these effects cannot be ruled out as a potential generic concern, especially when considering the PWR secondary water chemistry controls that have existed in the industry (see "Other Conditions" contributing to SCC).

Garigliano Steam Generator Event

The Garigliano steam generator crack developed at the inner surface of the water box circumferential weld between the tube sheet and the nozzles on the primary side (August 1978). The through-wall crack propagated through the Monel cladd, and the SA-302-B shell (approximately 2 inches thick). General Electric conducted an extensive investigation, reporting the results in Reference 10. The most pertinent information revealed that the crack propagation resulted from environmentally assisted corrosion under sustained loads (SCC). Manganese Sulfide as segregates were evident in the Monel and base metal, with the presence of Sulfur in the region of crack tips. Therefore, aggressive acidic crack-tip chemistry caused by dissolution of the sulfide inclusions were concluded by GE to be contributors to the SCC. Local post weld heat treatment (PWHT) of the weld with unknown control was also reported by GE to have resulted in high residual stresses in the region of the weld. The high oxygen content (~200 ppb) in the coolant medium was not considered atypical, but it may have enhanced the electrochemical reaction involved in the crack initiation and propagation.

GE concluded that the conditions that prevailed in the Garigliano steam generator (high residual stress, material sulfur content and inclusions) were atypical of current domestic BWR design and PWHTs. The NRC staff did not challenge the GE position. Therefore, the Garigliano event was not considered a generic event typical to domestic operating BWRs. However, the effects of sulfur content in the material and the potential contribution to SCC have since been subject to further tests and evaluations (see discussion on material testing). One might argue in hindsight that the Garigliano event could have been a precursor to the SCC susceptibility of high/low sulfur content low-alloy steels in reactor grade water.

HDR Rupture Event

NUREG-1061, Volume 3, (Reference 11) describes the double-end guillotine break that occurred in the HDR test facility on November 3, 1983. The reducer (conic section) that failed was fabricated from a single billet of 15 Mo 3 steel. The wall thickness of the conic section was approximately one-fourth the design thickness. Therefore, the combined primary, secondary, bending, and notch stress concentrations, could have resulted in a stress intensity of nearly two orders of magnitude above the design stress. This fabrication error could well have resulted in exceedance of some stress threshold that caused the failure. The thinness of the conic wall section, and the high oxygen content (~8ppm) may also have contributed to the failure.

The atypical design and fabrication errors related to the HDR failure are believed sufficient to preclude this event as representative support of this issue as a generic issue. In accordance with Reference 32, it must be pointed out that although the stresses were very high, there was no gross plastic deformation and no ductility exhibited on a micro scale. It was a brittle fracture. The failure is atypical of fatigue in that there were numerous initiation sites. These facts point to stress corrosion cracking of low alloy/carbon steels as the failure mechanism. This incident is cited to demonstrate the mechanism.

PWR Feedwater Line Cracking Events

These failures are being addressed in Generic Safety Issue 14. The primary failure mode has been identified as thermal fatigue (not CF or SCC) resulting from coolant stratification. The PWR Pipe Crack Study Group completed its investigation of this issue and published its findings in NUREG-0691 (Reference 12). Based on the above findings, any SCC that may, or may not have influenced the resulting failures were masked by the thermal fatigue constituent.

Other Events Contributing to Potential SSC

Intrusion of chloride, sulfide, copper, and other contaminants into the BWR reactor water and PWR secondary water may contribute to SCC of the vessels materials. A report, EPRI NP-1136 (Reference 13), stated that 20 BWR plants over a 33-month time period (1974-1977) indicated 12 forced outages as a result of high conductivity in the reactor water or heavy condenser tube leakages. On an average, this amounts to 0.22 significant contaminant intrusions per BWR reactor year. EPRI NP-2230 (Reference 14) reported 6 PWR plant condenser leakages over 172 plant years of operation. This amounts to a frequency of 0.03 contaminant intrusions from condenser leaks into the PWR secondary cooling water of the steam generators.

As a further example of other apparent poor PWR secondary water chemistry operations, in addition to the IP-3 sludge analyses discussed earlier, the sludge deposits in the removed Surry 2A steam generator undergoing tests at Hanford were reported in NUREG/CR-3842 (Reference 16). Analyses of the Surry sludge deposits revealed 35 to 60 percent metallic copper, 20 to 30 percent Hematite (Fe_2O_3), and 10 percent Cuprite (Cu_2O). All the analytical data on the sludge samples indicated that they originated from the secondary side. The high copper content probably originating from the condenser tubing (see "Other Considerations").

Tighter requirements for the BWR reactor water may account for the reported higher frequency of contaminant intrusion for BWRs from condenser tube leaks. However, Regulatory Guide 1.56 (Reference 15) provides methods determined acceptable by the NRC staff to maintain high purity water in the BWR reactor

water cycles and to minimize failure of the reactor vessel from mechanisms of general corrosion and SCC induced by impurities in the reactor coolant.

For the secondary side of the PWRs, resolution of USI A3, A4, and A5 (Reference 17) contained staff recommendations that the PWR plants incorporate Revision 3 to the Standard Review Plan (SRP) Section 5.4.2.1 (Reference 18) as plant-specific programs for secondary water chemistry control.

From the above limited data, condenser tube leaks in BWRs and PWRs have been frequent. However, the water purity requirements for BWR plants should alleviate potential corrosion effects to the BWR reactor vessels. For the PWR steam generators, adoption of the secondary water chemistry guidelines may reduce future corrosion potentials, but not necessarily resolve the effects of existing corrosion damage.

Based on the IP-3 experience (see previous discussion), the above described Surry sludge analyses, the recent Surry Unit 2 inspections discussed in "Other Considerations," and that steam generator tube degradations have been linked to variable PWR secondary water chemistry controls (Reference 3), it appears reasonable to equate the adequacy of the steam generator secondary water chemistry environment to conditions that may also enhance SCC in the steam generator vessel shells.

Inferences from Materials Testing

A considerable amount of materials research and testing has been performed on the SA-508 and SA-533 reactor vessel materials. The research and testing were performed in typical PWR and typical BWR reactor water chemistries. The research results (See References 19 to 27) also included comparisons with the ASME Section XI air and water fault lines. Based on the existing research results, the following generalizations appear appropriate for these materials: (1) there is a trend toward increased crack growth rate with higher material sulfur content, (2) a higher dissolved oxygen content results in higher initial crack growth rate, but the crack growth rate is stifled with crack depth such that after an initial period of crack growth rate the effects of the bulk solution dissolved oxygen content diminishes. Therefore, there is little difference in the effective crack growth rates of these materials in BWR and PWR reactor water chemistries, (3) the crack growth rates for reactor pressure vessel materials are within, or consistent with, the ASME Section XI surface (wet) fault lines.

The most significant effect observed was the high/low sulfur content (material variability), and not the oxygen content (environmental variability). The aqueous solutions used in the referenced research did not contain copper in solution, but some tests did contain small amounts of chlorine in solution.

The only research test results obtained for the SA-302-B base material and associated weld material is reported in References 7 and 8. These results were discussed in the earlier IP-3 comparisons.

Based on the above discussions, the differences in the dissolved oxygen contents for the BWR and PWR reactor water chemistries are estimated to have little or no effect on the probability of increased crack growth rates for the reactor pressure vessels. Only limited information was available for the (SA-302-B) pressure vessel material. In the presence of the simulated and degraded PWR secondary water chemistry, the SA-302 material may be susceptible to some form of accelerated corrosion attack.

Prioritization Analyses

BWR Frequency and Consequence Estimates

BWR Reactor Pressure Vessel Rupture Frequencies Estimates

A nominal base case pressure vessel rupture frequency of 1.0E-7/Ry is assumed reasonable for the BWR reactor vessels (Ref. 28). In consideration of the following: (1) research results of the reactor vessels materials, in their respective reactor water chemistry environments, the vessels materials crack growth rates are within the ASME code limits, (2) the protective corrosion shield provided by the cladding on the inside surface of the reactor vessels and, (3) the BWR reactor water chemistry requirements described earlier; no significant increase in the BWR reactor vessel rupture frequency from SCC is anticipated. However, to provide a coarse estimate, it is assumed that a 25 percent increase in the BWR reactor vessel rupture frequency can be attributed to SCC. This potential increase in BWR reactor vessel rupture frequency is based on the percentage of stainless steel pipe ruptures attributed to SCC reported in NUREG-1061, Volume 1 (Reference 11). Because of the observed prominence of SSC in stainless steel pipes, it seems unlikely that the percentage of reactor pressure vessel ruptures due to SCC would exceed 25 percent of the total vessel rupture frequency without prior history of this condition.

The change in BWR reactor pressure vessel rupture frequency that may be attributed to SCC, is therefore estimated at 2.5E-8/Ry.

BWR Consequences Estimate

Assuming that SCC provides a potential change in the BWR reactor vessel rupture frequency (2.5 E-8/Ry), the probabilities of radioactive releases in BWR categories 2 and 3, as described in WASH-1400, are 0.1 and 0.9, respectively. Assuming a 1120 MWe BWR, meteorology typical of the Braidwood site, and a surrounding uniform population density of 340 persons per square mile, the public radioactive risk within a 50 mile radius is 0.113 man rem per

reactor year. Considering a remaining reactor life of approximately 30 years, the public risk is 3.5 man rem per reactor.

PWR Frequency and Consequences Analyses

A steam generator leak or rupture of a single SG would likely produce a rapid cooldown of the reactor similar to an inadvertent full-opening of the turbine bypass valves or a main steam line break (Ref. 28). The containments are capable of sustaining a complete blowdown of a steam generator. Therefore, rupture of a single SG with no additional failures has no significant risk to the public from core melt or radioactive releases through containment failures. The plant operations, and operation responses to such an event are assumed similar to those described in Generic Safety Issue A-22 for a steamline break inside containment. In addition, subsequent and detailed staff evaluations on PWR responses to main steam line breaks (MSLB) with concurrent steam generator tube ruptures (SGTRs) and small-break LOCAs were reported in Reference 35. The conclusions reported in Ref. 35 were that a MSLB inside containment (similar to a SGR) would likely be bounded by the FSAR analyses and not result in a core melt.

For an SGR to lead to a significant release (core melt) the SGR must be accompanied by damage to the reactor coolant system (RCS) and failure of the emergency core cooling system (ECCS), or failure of the auxiliary feedwater system (AFWS) and the ECCS. The following sections will address these PWR systemic events.

PWR Steam Generator Rupture Frequency Estimates

WASH-1400 estimated that the steam generator (SG) rupture frequency was similar to the reactor pressure vessel (RPV) rupture frequency (1E-7 per year). Considering approximately 3 SGs per reactor, the base case SG rupture frequency is 3E-7/Ry.

To assess the potential increase in SG rupture frequency as a result of accelerated SCC or CF, from PWR secondary water chemistry variability between plants, we reason the following: (1) plants with clean secondary water chemistry will have a SG rupture frequency equal to the above base case rupture frequency (3E-7/Ry), (2) plants that have experienced medium degradations of the SG tubes will have a SG rupture frequency one order of magnitude greater (3E-6/Ry) than the base case, (3) plants that have experienced severe degradations of the SG tubes will have a SG rupture frequency two orders of magnitude (3E-5/Ry) greater than the base case rupture frequency of 3E-7/Ry.

The above SGR frequency (3E-5/RY) is back calculated to estimate the number of steam generator leaks that have occurred by using the piping leak-before-break (LBB) ratio of 20 from Reference 28. The predicted number of SGLs

based on the above reasoning is (3E-5 SGR/RY)(500 RY)(20 L/R) ~ 0.3 SGL. Likewise, if we estimate that the SG inservice inspections have a 10% chance of not detecting cracks in the steam generators before they develop into leaks, three (3) steam generators with cracks could be expected. Compared to the seven (7) steam generators where cracking has been detected, the above crude estimates are fairly good, but a better correlation with leaks and cracks would be obtained from a SGR frequency of 1E-4/RY. For comparative purposes, the probability of MSLB is also 1E-4/RY.

Alternately noting that no SGRs have occurred in 1500 SGY (500 RY) and ignoring the current SG ISI experiences for leak to crack detection (1/7) and LBB experiences in U.S. and foreign plants (2 leaks with no ruptures), we would estimate a SGR frequency of 1E-3/RY. The SGR frequency of IE-3/RY therefore represents a bounding, but prudent estimate. Ignoring the ISI crack detection capability and LBB experiences appears prudent because of the uncertainties in estimating these early warning indicators. As an example of the conservatism of ignoring the crack detection capability; a very conservative staff fracture mechanics analysis (Ref. 36) estimated that a catastrophic rupture of the steam generator would only be predicted to occur from a complete circumferential crack (360°), with a crack depth approaching one-half the SG vessel wall thickness. A crack of this magnitude seems very likely to be detectable. Therefore, the SGR frequency may range from a best estimate value of 1E-4/RY to an upper bound estimate of 1E-3/RY.

PWR Steam Generator Support (SGS) Failure and Loss-of-Coolant-Accident (LOCA) Frequencies

If cracks develop in the SG vessel shells, Ref. 28 and Ref. 34 independently judged that the SG would likely leak before rupture. The SGR event would therefore most likely be bounded by the MSLB event previously discussed. However, should a catastrophic SGR occur, the SG reaction loading to the steam generator support (SGS) structure is highly uncertain. In recognition of this, we will assume the conditional probability of 0.5 for failure for the SGS (SGS/SGR). The SGS/SGR = 0.5 infers that the SGS is as likely to fail as not to fail. Given failure of the SGS, we assume the conditional probability of a large break LOCA (LBLOCA), given a SGS failure, is 1.0.

PWR Core Melt Frequencies

The systemic events that are assumed to lead to core melt conditions as a result of a catastrohic SGR are damage to the RCS (LBLOCA) and failure of the emergency core cooling system (ECCS) in the unaffected loops, or failure of the auxiliary feedwater system (AFWS) and the ECCS in the unaffected loops. The estimated upper bound core melt frequencies for these sequences are as follows:

Failure Event	Failure Probability/RY
SGR	1E-3
SGS/SGR	5E-1
LBLOCA/SGS	1.0
ECCS Failure	1E-2
	5E-6

Failure Event	Probability/RY
SGR	1E-3
AFWS Failure	4E-5
ECCS Failure	1E-2
	4E-10 (negligible)

PWR Containment Failure Matrix

Containment response to a core melt accident from the above LBLOCA/SGR can be grouped into separate plant damage states (PDS). The plant damage state depends on the availability of equipment or systems to reduce containment temperature and pressure, and/or plant damage states involving containment bypass or failure to isolate containment. The PDS descriptions, and probabilities resulting from the LBLOCA/SGR are as follows:

Plant Damage State (PDS)

PDS	Description	Probability
A	No containment heat removal or containment sprays	1E-3 (Ref. 28)
B	Containment heat removal and containment sprays available	0.998
V/B	Given B, but containment bypass through failed MSIVs in ruptured SG steam line	1E-3 (Ref. 29)

The containment failure modes are similar to those used in WASH-1400 (Ref. 28). The conditional probability of the containment failure mode for each PDS is shown in the table below:

Conditional Containment Failure Mode[a]

PDS	α	δ	$\beta 4$	$\beta 5$	V
A	E-2	.96	E-2	-	-
B	E-2	-	-	E-2	-
V/B	-	-	-	-	E-3

(a) α, δ, β, V, are the containment failure mode conditional probabilities for missile damage, overpressurization, failure to isolate, and bypass, respectively.

The probability of an α failure mode (α=1E-2) from a SGR refers to direct containment failure by missile penetration. For a LBLOCA induced core melt, the in-reactor-vessel steam explosion has a probability of 1E-4 to produce a missile that breeches containment. For purposes of this analysis, the α failure mode probability from missiles generated by the SGR is assumed as 100 times greater than that from an in-reactor-vessel steam explosion. Therefore, even through an in-reactor-vessel steam explosion is likely to occur from a core melt down, its contribution to containment failure is negligible.The corresponding WASH-1400 α release category is a Category 1 release due to the containment failure by a missile generated by the SGR.

Steam produced by the SGR and reactor molten fuel (core-melt) with water in the reactor cavity can fail the containment by overpressurization (δ). This would occur only when containment cooling is lost (Ref. 28, 35). The probability of overpressurization due to hydrogen burn is assumed negligible because the steam concentration in containment will tend to suppress hydrogen burn propagation. The probability of the δ mode failures for PDSs A and B are assumed as 0.96 and zero, respectively. The corresponding WASH-1400 release for PDS A and B are Category 2 and Category 3, respectively.

Failure to isolate containment (β failure mode) is assumed to have a probability of 0.01. The $\beta 4$ mode is with containment sprays unavailable, and the $\beta 5$ mode is with containment sprays available. The corresponding WASH-1400 release categories for $\beta 4$ and $\beta 5$ are Category 4 and Category 5, respectively. The "V" failure mode probability of 0.001 (Ref. 29) represent containment bypass through the ruptured steam lines in the affected loop with the main steam line isolation valves (MSIVs) failed open. The conditional PDS=V/B assumes containment sprays are available, and the corresponding WASH-1400 release category is a Category 3 release.

The basemat melt-through failure mode is a relatively benign failure mode, and with the most likely case of the containment sprays being available, we assume basemat melt-through is precluded.

The large break LOCA (LBLOCA) assumed to be induced by the SGR may also be accompanied by steam generator tube ruptures (SGTRs) in the affected loop. However, the conditional SGTRs would be dominated by the probability and consequences of the LBLOCA sequences.

PWR Risk Sequences

The PWR risk consequences for a core-melt frequency (5E-6/RY) resulting from a SGR induced LBLOCA is 0.4 man rem per reactor year. Over a remaining plant life of 30 years the public risk is 12 man rem/R. The tabulations of the calculated public risk parameters are:

Public Risk Parameters

WASH-1400 Release Category	Containment Failure Mode	Release Frequency per RY	Conditional Dose (man-rem) per Release	Public Risk (man-rem/RY)
1	α	5E-8	5.4E+6	.3
2	δ	5E-9	4.8E+6	.02
3	V	5E-9	5.4E+6	.03
4	β4	5E-11	2.7E+6	-
5	β5	5E-8	1.0E+6	.05
Total	-	1E-7	-	.4

The release categories and corresponding containment failure modes are described in the Containment Matrix Section. The release frequencies (Column 3) are the products of the core-melt frequency (5E-6/RY) times the summed products of the PDS and the conditional containment failure mode probabilities for each PDS that are provided in the Containment Matrix Section. The conditional dose (Column 4) is the man-rem per release for each release category. These release doses are based on the fission product inventory of a 1120 MWe PWR, meteorology typical of the Byron site, and a surrounding uniform population density of 340 persons per square mile over a fifty mile radius from the plant site, with an exclusion radius of one-half mile from the plant.

Cost Estimates

Based on discussions with the RES Project Manager (A. Toboada) for Environmental-Assisted-Stress-Corrosion (Reference 30), this issue could be incorporated into the yet-to-be finalized long-term research plans at no

additional NRC cost. A near-term effort would involve an initial expenditure of NRC research funds ($265,000). Depending on the outcome of the research results, additional NRC and industry funds may be needed to develop a solution(s). Because of the small risk, no other costs were estimated.

The industry has a significant economic incentive to repair surface cracks in their steam generators - before they develop to through-wall cracks. As an example, repair of steam generator surface cracks at the Surry plant involved removal by grinding (repair welding was not necessary) at an MTEB estimated cost of approximately one million dollars. At the Indian Point plant, where a small through-wall crack developed in one steam generator, the repairs involved grinding and weld repairs. The MTEB estimated costs to the Indian Point facility was approximately $8 million. In neither of these cases were the plants required to go into a forced outage situation. However, should a plant be placed into a forced outage situation as a result of through-wall cracks in the steam generators, the average replacement power costs of approximately $500,000 per day, in addition to the repair costs, would likely result in costs well in excess of $8 million.

Other Considerations

Steam Generator Inspection Considerations

A comparison was made of the plants reported by Westinghouse (Reference 6) as having been inspected for indications similar to the IP-3 flaw with plants that have experienced severe SG tube degradation histories (Reference 3). The comparison indicated that, in general, the plants inspected were not plants with histories of severe steam generator tube degradations. Subsequent inspections of the replaced Surry Unit 2 steam generators have revealed intermittent cracks up to a quarter-inch deep (Ref. 31). The cracks were in the transition region that was part of the original steam generator. The transition cone wall thickness in this area is 3.4 inches thick, and is required by design to be at least 2.0 inches thick. Because these indications were in the original part of the transition cone, the affected material was exposed to the same poor secondary water chemistry discussed earlier. The cracking of three Surry 2 steam generator shells occurred at the same joint as the four Indian Point 3 steam generator shells. The inspections of the joints have predominantly been by ultrasonic testing methods from the outside of the shell. As experienced in some of the BWR stainless steel piping inspections for stress corrosion cracking, the ultrasonic testing indications were incorrectly ascribed to geometric configuration. In this regard, IE Information Notice 85-65 (Ref. 33) has informed the industry of the events at Indian Point and Surry, and the experience with ultrasonic versus magnetic particle examinations related to crack detection in the steam generators. Therefore, subsequent ISI testing of the SGS should be more reliable and thereby further reduce the chance of a SGR.

Conclusions

Based on limited operating experiences (one steam generator leak in U.S. domestic plants, and one steam generator leak in foreign plants) and expert opinion (Ref. 34), the steam generators outer shells are more likely to leak than to catastrophically rupture. A significant leak in a steam generator outer shell would be expected to result in plant responses comparable to a transient induced by the inadvertent full-opening of the turbine bypass valves. A larger steam generator leak (small rupture) is expected to be bounded by the MSLB with concurrent SGTRs and SBLOCA as evaluated in Ref. 35. The Ref. 35 detailed analyses determined that such an event would not result in a core-melt accident.

To further bound the probability and consequences of this issue we have ignored the SG crack detection experiences and SG leak experiences (that essentially have provided defense-in-depth mitigations to severe SG ruptures) and assumed a catastrophic SGR probability of 1E-3/RY that leads to a LBLOCA (failure of primary piping loop). Based on this scenario as a bounding analysis, the public risk from a SGR was estimated at 12 man-rem/PWR plant. Therefore, the risk reduction potential for this issue (3.5 man-rem/BWR plant, 12 man-rem/PWR plant) indicate that this issue is of low safety significance to the public.

The quantified valves used in this evaluation contain a number of unquantified uncertainties. However, to the extents judged reasonable, the bounding values are believed to be biased in conservative directions. Thus, these estimates are more sensitivity studies than absolute quantifications and, therefore only represent the potential safety significance of this issue relative to other issues.

We have also considered other concerns raised by the Materials Engineering Branch (Ref. 32): "the experience at two plants (Indian Point 3 and Surry 2) of the material failure mechanism that was not addressed in the original design (and raised doubt whether General Design Criterion 4 is being met) requires a response by the staff. The research effort promised in the future would be too late to address licensing concerns now, especially for operating plants. Active consideration should be given to placing a higher priority on research efforts to enhance our understanding in order to provide a meaningful, timely response." However, MTEB also concluded (Ref. 32) that this issue only provides a minimal risk to the public health and safety in terms of the contribution to core melt probability.

Based on 1) the low public risks for this issue, 2) the MTEB expert opinion that SG leaks are more likely than SGRs (Ref. 34) - which is currently supported by the Indian Point and Garigliano experiences, 3) existing staff recommendations to the industry to implement improved secondary water chemistry programs (Ref. 17), 4) the IE Information Notice (Ref. 33) that

should promote more reliable SG inspections, and 5) the industry economic incentive to resolve this issue; this issue has minimal public risk, that will be even further reduced by implementation of the above actions.

However, the MTEB concerns related to the need for a better understanding of the materials cracking morphology, potential licensing position(s) related to meeting the original licensing design bases, and whether or not the General Design Criteria are met, are considered licensing concerns.

Therefore, based on the above evaluations, staff actions already taken, and the above discussions, we recommend that this issue be classified as a Licensing Issue and not a safety issue.

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