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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 16, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Dr. Joram Hopenfeld
Engineering Research Applications Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY ISSUES

In response to your November 1, 1999 request regarding the final staff Differing Professional Opinion Consideration Document (DPO Consideration), undated, I must state that none of the DPO issues has been resolved to my satisfaction. The DPO Consideration regards these issues as not being serious public safety concerns. It misstates material facts, ignores major DPO documents, and focuses on minor issues instead of addressing all concerns in an objective and professional manner. Some of the models proposed by the staff to refute the DPO issues border on fiction. This and the attached memorandum address the reasons for my disagreement with the DPO Consideration and certain of your November 1, 1999 comments. For the convenience of the ad hoc panel I have attached supporting documents to the DPO Reply Document (DPO Reply). The ad hoc panel should consider all the supporting material, Attachments 1-8, as an integral part of the DPO Reply.

The DPO Reply shows conclusively that the NRC cannot technically justify replacing historical deterministic regulations based on engineering determinations with risk-informed regulations that use very sparse data and/or are dominated by inscrutable and subjective Bayesian priors. To be credible, risk-informed regulation mandates statistically valid and scrutable data, competent insights of accident scenarios and their consequences, and of accident prevention strategies, as well as meaningful public involvement. In reality, the staff examines accident scenarios and their consequences in a superficial manner; accident prevention is apparently dictated primarily by financial considerations, and the public is being excluded from meaningful participation in the NRC deliberation process. This situation is exemplified in the recent granting of an inspection waiver to Farley Nuclear Power Plant Unit 1. Considering that "staff beliefs" were used as a sole justification, the inspection waiver shows that public risk from aging power plants has never been greater.

In your memorandum of November 1, 1999, the NRC hypothesis is that steam cutting of adjacent tubes causing a major failure of numerous tubes is not realistic for short cracks and that the Office of Research (RES) will conduct research to better quantify crack length dependence for steam generator tube failures. Your memorandum further stated that (1) the low frequency of severe accident sequences and (2) the limited duration of the Farley Nuclear Plant Unit 1 request, assure adequate protection of public health and safety.

I do not agree with your assessment as I show conclusively in the DPO Reply. For example, there is nothing in the engineering scaling laws of jet erosion to even remotely suggest that cracks shorter than 0.12" will not damage adjacent tubes. The fact is that 0.010" diameter jets are used in metal machining. Jet velocities, particle concentrations, and hardness are the main parameters affecting jet cutting.

The DPO has shown that a potential for jet cutting exists under both design bases and severe accidents. These results were the determining factor for arriving at a large early release frequency (LERF) of 10^{-4}/ry. Subsequently, in 1997, the staff also calculated erosion rates which could cut adjacent tubes within a few seconds. In the case of Farley, the staff used "their beliefs" as a basis to conclude that jet cutting would not occur and that, therefore,mid-cycle inspection could be waived.

When public safety is concerned and calculations show a potential for severe consequences, it is insufficient, in my opinion, to make operational decisions and rely on future tests even if such tests could show that very small through-wall cracks will not induce failure of adjacent tubes. The proposed tests are impractical, considering the uncertainties in abrasive particle loadings, particle shape, size, and composition. To take an indefensible non-conservative position ignoring well known and generally accepted engineering concepts is incomprehensible.

The statement in your November 1, 1999, memorandum states that the DPO has not shown that failure will occur when the cracks are extremely short; this is untrue. In my March 5, 1997, presentation to the ACRS, I clearly showed that a 0.001" hole can result in extremely large primary to secondary leakage before the surge line fails. Initially the staff excluded the possibility of short cracks (less than 0.25"). Since it was identified to the staff that this assumption is incorrect, the staff now claims that crack length, and not merely the presence of a crack, places a limit on damage potential. Such invention of theories to prove desired outcomes prevails throughout the DPO Consideration.

The staff, in addition to improperly assessing the severe accident scenario for the Farley first-of-a-kind risk-informed review, entirely ignored jet cutting during design basis accidents. In its supporting documentation (NUREG-1570), the staff stated that such jets will be quenched by water during design basis accidents. Only minimal engineering knowledge of accident phenomena is required to recognize that, following steam line break accidents, the tubes will be uncovered instantaneously and water will not be available to quench the jet. As shown in the DPO Reply, impingement of high velocity superheated droplets and aerosols could damage adjacent tubes.

The NRC's Farley review is a dangerous precedent for the new risk-informed policy. It is based on subjective judgement, selective use of data, and the proposition that if an accident did not happen in the past it will not happen in the near future.

I do not agree with your assessment that the Farley risk-informed review is a positive step. As discussed in the DPO Reply, the entire voltage-based methodology that was used for Farley's tube repair criteria is not capable of predicting accident leakages and of meeting Part 100 limits. The claim that the risk of tube- to- tube failure propagation is limited by low frequency events is irrelevant. The frequency of the initiating events has already been included in the calculations of the core melt frequency. I and the Office of Research independently have calculated that the large early release frequency will exceed the commission safety standard by an order of magnitude. This value, 10^{-4}/ry, was derived from analysis of several accident scenarios and is significantly larger than staff estimates. The staff value of $2x10^{-6}$ /ry was obtained from unrealistically simplified theoretical models and subjective judgements. This difference in the estimates of early dose releases is the basis for my concern that the NRC inspection waiver for Farley blatantly disregards public safety.

Your November 1, 1999 memo statement that "technical criteria which have been applied in the past will be applied to license renewals" raises a serious concern. The DPO shows that aging components must be treated differently from new components because the severity of the consequences from incidents caused by aged components is much higher.

Using an oversimplified mathematical model and without providing any justification for the underlying assumptions, the staff has been claiming that there are no safety significant differences in the behavior of aged and non-aged tubes during accidents (Draft NUREG-1477, "Voltage-Based Interim Plugging Criteria For Steam Generator Tubes", June 1993). These claims have not been substantiated; only a meaningful resolution of the DPO and GSI-163 can test their validity.

The design of the present generation of nuclear plants was based upon the assumption that steam generator tubes maintained their original strength during design basis accidents. Plant operating experience has demonstrated that this assumption is not valid for aged components. A design basis steam line break accident which assumes that the tubes are in their "as received" condition could result in a tube rupture, but it could lead to a multi-tube leakage if the tubes are aged and contain wall cracks. In the first case, the consequences to the public are minor, while in the second case they could be catastrophic. The level of risk of operating degraded steam generators such as those at Farley is patently unacceptable.

Your memo indicates that the NRC and the industry are now planning revisions to the existing regulatory framework for steam generators. You stated that closure of Generic Safety Issue GSI-163 will be based on the final resolution of the DPO. The procedure for resolving DPOs and GSIs are entirely different; GSIs are Congressionally mandated, a formal action plan including cost-benefit studies of alternative fixes is required. GSIs are subject to stricter reviews and public scrutiny than DPOs. I do not agree with your position that the closure of GSI-163 must await the resolution of the DPO. GSI-163 should be evaluated on its own merit and, consistent with mandated requirements and Advisory Committee on Reactor Safety (ACRS) recommendations, before the technical specifications are finalized. GSI 163, Multiple Steam Generator Tube Leakage, was assigned HIGH priority because the Office of Research determined that the probability of a major accident was $3.4x\ 10^{-4}$/ry if degraded tubes were allowed to remain in service.

The DPO was initiated in 1991 because the NRC had begun allowing plants to operate with through-wall cracks in steam generator tubes. I felt that the NRC failed to recognize the fact that leaving cracked tubes in service could, during design and severe accidents, result in primary to secondary leakage which would exceed the leakage from a single tube rupture. The plants were not designed for such large leakages and therefore public safety was compromised. After delays, the Office of Research reached the same conclusion, initiated GSI-163, Multiple Steam Generator Tube Leakage, and prioritized it as HIGH. In 1992 the Trojan reactor developed large leaks because degraded tubes were allowed to remain in service. In 1993, Congressman DeFazio, feeling misled by NRC management, called a meeting with the staff because he was not informed that a Differing Professional View (DPV) on this subject existed. The Congressman was assured that all the DPV issues were being taken into considerations in the preparation of a new Generic Letter,GL-95-05, "Steam Generator Tube Integrity." Nothing was further from the truth.

The DPV was ignored and the staff adopted the Westinghouse methodology relating to steam generator tube degradation in its entirety. Westinghouse at that time was in litigation in connection with the premature degradation of the tubes in their steam generators. In July 1994, I filed a DPO as a continuation of the less formal DPV process, about which the NRC had done essentially nothing in three years.

The NRC did not convene an ad hoc panel, as required by existing procedures in order to determine whether there is an immediate public risk. Instead, the NRC decided in 1993 to resolve the DPO issues as part of steam generator rulemaking, which typically would not be finalized for 5 or more years. When the rulemaking activities failed in 1997, the NRC decided to incorporate the DPO into Generic Letter activities. When the Generic Letter activities failed in 1998, the NRC decided to incorporate the DPO into the Regulatory Guide activities which failed following industry comments in July 1999. All these activities are now being replaced by discussions with the Nuclear Energy Institute (NEI) and the industry regarding industry initiative, NEI 97-06. The DPO concerns are not meaningfully considered in these discussions.

In 1997, the staff informed the Commission that they had recently discovered that the replacement of the 40% through-wall plugging criteria would significantly increase susceptibility to tube failure during certain severe accident sequences. Since the introduction of requirements to evaluate this phenomenon would have constituted a backfit, the staff informed the Commission that it was dropping the rule making activities. Conspicuously, the staff failed to inform the Commission that, five years earlier in 1992, a DPV analysis already existed which showed that lifting the 40% plugging criteria would significantly increase the risk from severe accidents. The staff knew or should have known that such an analysis already existed.

During the1995-1999 period, the NRC granted 17 reactor units relief from the 40% plugging rule under GL 95-05. In August 1999, using for the first time a risk-informed approach, the NRC granted Farley Nuclear Plant Unit 1 relief from steam generator tube inspection. As already mentioned above, "staff beliefs" was the basis for providing this inspection relief. When the Farley action was taken, the staff was aware of the DPO analysis showing that Farley could be susceptible to large primary leakage, which could exceed regulatory limits and endanger the health and safety of the public during both design basis and severe accidents.

It should be noted that in 1994 the ACRS endorsed GL 95-05 as an *"interim"* measure and did not consider the staff LERF of $2x10^{-6}$/ry to be conservative for steam line breaks; it also requested that the staff quantify the margins in the radiological dose estimates. The NRC staff presentations to the public, the Commission, and the ACRS, were all predicated on the assumption that GL 95-05 was an *"interim"* measure. It is apparent now that the staff is installing GL-95-05 permanently with outstanding issues.

The significant waste of NRC's limited resources could have been prevented if the NRC DPO and GSI procedures had not been ignored by the staff. The backfit issue should have been identified in 1993 and a resolution planned accordingly.

The staff claim that the DPO issues are continuously being addressed must be seriously questioned. The list of references provided in the DPO Consideration does not even include the analysis and conclusions for the DPV phase of the DPO/DPV process. Attachments 4 and 4A present the DPV analysis.

Recirculating-type steam generators used in Westinghouse and Combustion Engineering nuclear power plants are a monument to poor engineering. The plants are coping with aging steam generators by replacing them long before the end of their design life. Because of the long lead time and the unpredictability of steam generator tube aging, operators are faced with the choice of increasing the inspection frequency of steam generator tubes and suffering the financial consequences, or hoping that steam generator tube failure accidents will not occur until the steam generators are replaced.

Nuclear plant operators focus attention first on near-term needs. The public, on the other hand, has a long-term interest in having a sustained confidence that the NRC is using sound standards, valid and scrutable statistical data and analyses, adequate research, and sound engineering to protect their safety. Steam generator failure is one of the most serious accidents because of the potential for core melt and containment bypass. The Farley inspection waiver, the NRC's blanket adaptation of the Westinghouse methodology, the failure of NRC rulemaking and Generic Letter activities, and the eight years that the DPV/DPO have continued without closure are the best indicators as to where NRC priorities are, and have been for the past eight years.

The NRC and the NEI are now discussing changes to the present regulatory framework. The NEI comments (June 29, 1999) reflect the industry's desire to maintain maximum flexibility in repairing or replacing aging components, maintaining the same level of surveillance as in the past, and concerns of exposure to financial liabilities. The issue of leakage propagation from cracked tubes during design basis accidents and the uncertainties in meeting Part 100 limits are not addressed. The issue of reactor safety performance and how it can be measured when degraded components are left in service is also not discussed.

The key to risk- informed regulation is accountability to the public through valid and scrutable performance measurements. Since reactor accidents are low frequency and very high consequence events, measuring a reduction in plant accidents is not a practical benchmark. Reduction in relevant accident precursors and a demonstration that the NRC understands the root causes of potential accidents and takes relevant and timely preventive actions are quantities that can be measured or assessed. The deflection of public concerns behind a veil of obscurity, the development of easily tunable complex computer codes, the selective use of data, and the use of "staff beliefs" as an excuse for ignorance are examples of low score efforts toward gaining public credibility. Reduction in unscheduled outages, data acquisition for performance monitoring, well documented and publicly available NRC deliberations and publications in peer reviewed journals, well written standards, and meaningful consideration to public concerns are examples of high scores.

Your comment (November 1, 1999, memo) "as discussed in the DPO Consideration Document, existing steam generator programs related to steam generator tube integrity are adequate to insure public health and safety", and the statement that the DPO would not affect changes to steam generator technical specifications repudiate the entire DPO process. The purpose of the DPO is to first determine whether a safety problem exists and then to take the necessary actions to correct the situation. It is obvious that without even starting the DPO process with an independent review, you have predetermined the outcome. I would like to know the reason for this conclusion since the ACRS, as far back as 1994, had not endorsed the Voltage Based methodology for steam line break accidents. The ACRS considered the methodology on an interim basis only, this interim period has long expired.

The Executive Director for Operations has been assuring the Commission, the ACRS and the public that the DPV/DPO will be addressed as part of the regulatory approach for solving steam generator tube integrity issues. For nine years this has been the excuse given for not resolving the DPV/DPO in accordance with established procedures. Your statements reflect a position that you have decided without meaningfully considering all necessary facts, that the DPO issues will have no impact on the new regulatory framework. A memorandum from the ACRS to the EDO dated November 20, 1996 states: "Both the DPO and the GSI are directly related to the proposed rule making. We urge the staff to prepare a point by point response to the DPO and to prioritize and resolve GSI-163 before implementing the steam generator integrity rule." There has been absolutely no meaningful progress in resolving the DPO and the GSI since the above ACRS recommendation.

Your November 1, 1999 memo did not reply to my September 28, 1999, request for an ad hoc panel from outside the agency to review the DPO Consideration and the DPO Reply documents. A key provision of NRC Management Directive 10.159 is that a review of DPV/DPOs is to ensure "full consideration and prompt disposition of DPVs and DPOs by affording an independent impartial review by qualified personnel". Since the present DPV/DPO process has been under "consideration" for nine years, obviously the intent of "prompt disposition" has not been met.

The majority of internal NRC ad hoc panel members are typically appointed by NRC management. Should the panel find that the DPO issues are valid, it would be tantamount to stating that the NRC has been improperly placing higher priority on the nuclear industry's interests than on protecting public health and safety. Experience with internal NRC DPV/DPO ad hoc panels shows that such panels will not admit management wrongs no matter what the facts are.

Experience to date does not indicate that an internal NRC ad hoc panel will validly resolve the DPO issues; at best, it will set the stage for "additional studies". Your reference to future research on "jet cutting crack length dependence" indicates that the DPO already is being used as an excuse for further delay of closing this safety issue until all PWR plants have replaced their steam generators. The seven year NRC effort on steam generator related issues has unnecessarily expended significant NRC resources which could have been more effectively utilized for other generic safety issues. Additional waste would be unconscionable.

I therefore recommend that a panel from outside the agency, consisting of experts who have no link to the nuclear industry, and representatives for the general public, be selected to provide to the agency badly needed fresh ideas on how to deal with aging reactor components.

Please approve the release of this document to the Public Document Room (PDR).

cc: Chairman Meserve w/o atts
Commissioner Diaz w/o atts
Commissioner McGaffigan w/o atts
Commissioner Merrifield w/o atts
Sher Bahadur w/o atts

ATTACHMENTS

1. J. Hopenfeld, "Reply to The DPO Consideration Document" December 15, 1999.

Attachment 1- T.S. Kress to I. Selin, "Proposed Generic Letter 94-xx, "Voltage Repair Criteria For Westinghouse Steam Generator Tubes" ACRS Letter, September 12, 1994.

Attachment 2- R.L.Seale to S.A. Jackson, "Summary Report- Four Hundred Fortieth Meeting of the ACRS Commitee in Reactor safeguards..." Letter, October 21, 1997

Attachment 3- J. Hopenfeld Comments on the Thermal Hydraulic Analysis in NUREG-1570, ACRS Materials and Met. Subcommittee & Severe Accidents Subcommittee, March 5, 1997.

Attachment 4- Memo, Differing Professional View, December 23, 1991 and March 27, 1992

Attachment 4A - Memo, J. Hopenfeld to E. Beckjord, "Addendum to March 27, 1992 ,Memo Regarding Degraded Steam Generator Tubes," Sept. 11, 1992.

Attachment 5- Memo, J. Hopenfeld to W.D.Travers, "DPO Panel Review of Steam Generator Integrity," Sept. 28, 1999.

Attachment 6- J.Hopenfeld " Differing Professional Opinion Regarding NRC Approach to Steam Generator Aging," Sept. 25, 1998.

Attachment 7-Memo, J.Hopenfeld to J. T. Larkins "New Information Relative to Steam Generator Behavior During Severe Accidents," May 20, 1998.

Attachment 8- Memo, J.Hopenfeld to J.M. Taylor, "Differing Professional Opinion Regarding Voltage-Based Interim Repair Criteria for Steam Generator Tubes,". July 13, 1994.

REPLY TO THE DIFFERING PROFESSIONAL OPINION, (DPO), CONSIDERATION DOCUMENT

SUMMARY

The subject document is predicated on assumptions which show that Commission safety goals and Part 100 limits can be met when degraded steam generator tubes are allowed to remain in service. These assumptions presume very small primary/secondary leakages during postulated accident scenarios. When the leakage is small, the operator can typically control the release of radioactivity without harming the public. Using alternate models, this document shows that staff's assumptions have no technical justification and that the above conclusions involve large uncertainties.

In a typical aging steam generator, thousands of tight cracks could break through tube walls. Even one crack can result in a jet that would cut through the adjacent tube and propagate the damage to other tubes. Such propagation has been observed in industrial steam generators. The DPO analysis used steam/ liquid jets and steam/particle jets to calculate damage to adjacent tubes. Data on erosion rates from steam/liquid jets were based on measurements from ten different power plants, data on steam/particle jets were obtained on testing alloy 600 at a coal gasification plant. Vast amount of literature on jet erosion in machining and pipe cleaning support the findings that jet erosion is a potential source for tube to tube damage propagation. In addition, experiments show that cracks may plug and unplug with corrosion products, in an unpredicted manner. There is no significant difference between jet erosion potential during main steam line break and during station blackout accidents. The staff's conclusions that Commission goals and Part 100 limits can be met rests mainly on the unrealistic assumption that maximum leakage is only a function of pressure differential. The DPO concluded that tubes with through the wall cracks or are suspected of developing such cracks by the end of an operating cycle must be removed from service.

Following are the assumptions which are at issue.

- Leakage is a unique function of voltage as measured by eddy current probes.
- Historical plant data provides the basis for predicting crack growth rates during each cycle.
- Leakage during Main Steam Line Break (MS LB) accidents depends only on the pressure across the crack.

- The steam generator vessel remains filled with water throughout the MSLB accident.
- Support plates will prevent cracks enlargement during Station Blackout Accidents (SBO)
- The 500 iodine spiking factor assumed in dose calculations assures that predicted dose releases will not exceed Part 100 limits during MSLB accidents.
- Westinghouse 1/7 inlet plenum flow mixing tests can be used to calculate mixing in steam generators during Station Blackout Accidents .

1. NDE Issue

The concern is that existing Non Destructive Examination (NDE) techniques are inadequate to permit predictions of primary to secondary leakage during design basis events such as main steam line breaks (MSLBs).

1.1 Staff Position

The staff states that it agrees that eddy current testing (ECT) continues to be poor for determining the size of intergrannular stress corrosion cracking (IGSCC) cracks. Predictions of end of cycle (EOC) voltage distribution are reasonable although they fail at times to predict the maximum voltage. Predicted MSLB accident leakages, which are based on the predicted voltages are considered to be conservative.

Previous contention that the DPO issues were fully addressed and were reviewed by the Advisory Committee on Reactor Safety (ACRS) in connection with GL95-05 are repeated. It is indicated that the staff is working on this problem with the industry.

1.2 DPO Position

The staff response to the DPO issue is inadequate, it does not provide data to support its claim that the voltage based methods can be used to predict accident leakage. The Voltage based methodology has no scientific basis, it was invented by Westinghouse to provide a rationale for allowing plants to operate defective steam generators. The ACRS did review the staff response to the DPO but found it wanting.

1.3 DISCUSSION

Instead of relying on analysis the staff uses "professional judgement" and purported ACRS endorsement as a justification for granting licensees relief from the 40% plugging rule. ACRS letters to the Commission Sept. 12, 1994 and Oct 1997, (Attachments 1 and 2) show that staff claims of ACRS endorsement are not true. Attachment 1 states that the low CDF value derived by the staff are not applicable for MSLB and Attachment 2 states that the "Committee plans to review the proposed final resolution of the issues in the DPO." Nowhere does it state that the DPO issues were satisfactory resolved. Staff presentations to the ACRS in 1994 stressed that GL95-05 was only as an interim measure , soon to be replaced by a new steam generator rule. The release of GL95-05 for public comments was also implied that this was an interim measure. In Attachment 1, the ACRS clearly indicated that GL95-05 was an interim measure and did not apply to MSLB's. The effectiveness of checks and balances at the NRC can be judged by the fact that the staff is now approving the voltage based SG repair criteria, per GL95-05, on a permanent bases to MSLB's.

Five years of expensive steam generator rule making activities ended when the staff "discovered" that there were potential problems with operating degraded tubes under severe accident conditions. It is important to note that the severe accident problem was identified by the DPV in 1992, Attachment 4A. If the DPV was meaningfully addressed at that time, the backfit issue could have been identified at a much earlier time. Alternative fixes and a more meaningful cost benefit analysis could have been conducted for backfit considerations.

The DPO Consideration document states that additional guidelines are being developed concerning flaw detection and measurements. As discussed below, no amount of staff work can change the fact that the measured quantity (voltage) does not correlate with primary to secondary leakage during postulated accidents and historical data on stress corrosion can not be used to predict future corrosion rates.

Why Voltage Is not Related to Leakage. The voltage produced by the eddy current probe is related to crack volume in its field of view. In contrast, leakage depends on the cross sectional area of those cracks which penetrated the tube wall. One may have a large voltage without any leakage if many crack are present below the surface. On the other hand a very small voltage may produce a large leakage if a single crack penetrated the surface. Therefore, voltage is not a unique function of leakage. The understanding of this point is important in the analysis and interpretation of laboratory data. If a sufficiently large number of samples is tested in a laboratory, a certain number will exhibit correlation between voltage and leakage. However, because of the non-unique relation between voltage and leakage, application of this data can not be used beyond the laboratory environment. The use of such data without a thorough analysis of uncertainties is not justified.

The space between the tube and tube support structure forms a crevice (a flow starved region) for the accumulation of chemicals which are left behind when the water in the crevice evaporates. Wetting and un-wetting inside the crevice results in a highly concentrated water solutions. Due to variation in crevice size and tube-to-tube heat transfer, large variations in chemistry exists in the steam generator. Since stress corrosion cracking is controlled by the specific chemistry and the surface stress of the tube, and since it is not possible to measure this parameter, laboratory tests inherently contain large uncertainties which can not be correlated statistically. The laboratory generated data base includes some samples of failed tubes which were removed from service. The chemical environment and degree of deformation which these samples had undergone during removal is unknown.

Why historical Crack Behavior can not be used to predict future crack growth. It is commonly accepted that stress corrosion cracking is a complex process defying predictions. Laboratory tests can be used only to screen different materials because crack formation and growth is controlled by numerous electrochemical, metallurgical and stress variables. Crack growth at a given stress level, depends on the conductivity and therefore on the concentration of the various species within the crack. Since local flow and chemical transients vary in service in an unpredictable manner, crack growth rates and crack topography also vary from cycle to cycle. Operating experience clearly demonstrates that past crack growth rates are not conservative with respect to future crack behavior. Farley steam generator -data show that crack growth rates for cycle 14 were higher than those for the two prior cycles. A voltage reading of 13.7 was measured, significantly above the predicted peak voltage of 7.6. During an MSLB accident, the presence of even one through the wall crack can cascade the accident as shown in attachment 3.

Stress corrosion can be characterized as a two step process; initiation and growth. Once a crack has been initiated, its mode and rate of propagation is governed by the local stress and chemistry. Even when the cracks propagate slowly, or are in arrest, other cracks are being initiated interacting with previous cracks to form a complex network which ultimately defines the tube strength. Cycle to cycle voltage measurements reveal nothing about crack arrest and growth cycles.

Because of considerable industrial experience on stress corrosion the common engineering practice is not to operate with components susceptible to stress corrosion cracking.

2. MSLB Leakage Issue

2.1 Staff Position

The staff position is based on the claims that the maximum primary to secondary leakage is limited by Emergency Core Cooling System (ECCS) pump capability and that the leakage depends only on the number of pre-existing defects, their size, and the maximum pressure during the accident.

2.2 DPO Position

The staff assumptions result in very small leakages, typically less than 100gpm. These assumptions ignore the poor sensitivity of the NDE technique and jet erosion of adjacent tubes and crack plugging. The primary-to-secondary leakage following MSLBs is expected to be significantly higher than 100 gpm because leakage does not depend only on the number of pre-existing defects. Even one defect could result in an area which could allow a flow of several thousands gpms The claim that the maximum leakage depends on ECCS pump capability is a direct result of the assumption that leakage coincides with crack enlargement from pressure loads alone. This assumption is hidden in NUREG-1477; and its implication on uncertainties is not mentioned. NUREG-1477, provides no justification that leakage will be limited to 100gpm.

This issue remains unresolved, as shown in Attachment 4 the RWST will be depleted following MSLB and 10CFR Part 100 will not be met.

2.3 Discussion

My presentation to the ACRS in 1997, Attachment 3, and my memo dated March 1992, Attachment 4, show that in addition to crack enlargement due to pressure, jet erosion and sudden unplugging of pre-existing cracks could also increase the flow area during accidents. Erosion calculations in Attachment 4 were based on droplet impingement alone free from aerosols. These calculations show that jet penetration time of an adjacent tube was on the same time scale that the steam generator maybe empty following MSLBs. This rationale leads to the prediction of large accident leakages and several thousand gpms which would prevent the operator from taking corrective actions. Following my findings, the NRC Office of Research (RES) concluded from the Trojan degraded SG tubes that flow rates as high as 1350 GPM were possible. For flow rates larger than 1000 gpm, the ability of an operator to control the accident and prevent exhausting the refueling water storage tank (RWST) and core melt must be seriously assessed using simulators. Instead, the DPO Consideration relies only on an unrealistic leakage model which was published in a draft form in NUREG-1477. The fallacy of this model must be clearly understood, because the staff habitually states the results without disclosing their underlying assumptions.

Why Leakage During MSLBs Does not Depend on Pressure. Draft NUREG-1477 assumes that maximum leakage coincides with crack enlargement from pressure loads during the MSLB accident. Eddy current voltage signals from in-service inspection and historical crack growth data are used to predict the number of leaking cracks during the accident.

Even if one was willing to assume that the present NDE techniques could identify all the preexisting cracks, it is important to realize that pressure loads alone are not the only source for flow area enlargement. Jet erosion and sudden unplugging of cracks could lead to a more significant area enlargement than pressure loads.

The jet emerging from a single through the wall crack contains small water droplets and micron size particle which impact adjacent tubes at very high velocities. Such jets (1) are known to initiate and propagate large leaks in conventional boilers (2) to damage turbine blades, and (3) are used for machining hard metals. The steam generator environment during MSLB and SBO accidents would allow a single jet to rapidly enlarge tube leakage, including by impinging on adjacent tubes. Attachments 3 and 4 show that the erosion rates from jets, obtained from different data sources, are sufficiently rapid to propagate catastrophic tube leakage increase.

Cracks are normally filled with chemical deposits which can plug, limit, or prevent tube leakage under steady operating conditions. Pressure or thermal transients however, may dislodge those deposits, causing an abrupt increase in tube leakage. As shown in PNL-4008, sudden leakage changes through steam generator tube cracks can occur in an unpredictable and random manner.

The model in NUREG-1477 generates an artificial and unsupportable upper limit to accident leakage. The conclusion that the amount of water lost through a ruptured area is balanced by the water that is injected by the ECCS is valid only for hypothetical cases where jet erosion and unplugging of cracks can be ignored. Attachments 4 and 4A, discuss some of these issues in more detail, as can be seen from the reference list in the DPO Consideration this work was neither recognized or acknowledged. Unlike NUREG-1477 which assume 99.9% operator success in controlling the accident, Attachment 4 assumes no operator action for most cases studied. The complexity of the accident and the fact that the systems were not designed for high primary to secondary leaks combined with a steam line break does not justify high credit for operator action. In one study case, where certain operator actions were allowed the RWST was still depleted.

3. Risk Increase Issue

The concern is that the large early release frequency, (LERF) will exceed the Commission Safety guideline of 10^{-5} per reactor year.

3.1 Staff Position

The impact of tube leakage induced by steam generator secondary side depressurization on the core damage frequency (CDF) and LERF are $2x10^{-6}$ and $3.9x10^{-6}$ per reactor year respectively.

3.2 DPO Position

The LERF is 10^{-4} per reactor year because of two reasons: (1) Once a leak is initiated following a steam line break, a stuck open relief valve, or station blackout primary to secondary leakage will propagate from tube to tube and increase in magnitude, and (2) above 1000 gpm the operator may not be able to bring the system to mid-loop operation and stop the leakage before the RWST is exhausted.

3.3 Discussion

3.3.1 Design Bases Accidents. As discussed above, Attachments 3 and 4 illustrate that steam jets, with or without particles, can rapidly damage adjacent tubes. In NURG-1570, the staff also calculated that it would take 4.9 seconds for a jet from a failed steam generator tube to propagate the failure to an adjacent tube. To deny the DPO position that such jets could lead to a very large primary to secondary leakage, the staff argues that the subcooled water in the primary side and the presence of water in the secondary side will quench the jet. This ignores the fact that during an MSLB or stuck open relief valve accident a recirculating steam generator becomes dry instantaneously, it takes about 20 minutes to fill it up and than parts of tubes are uncovered again due to boil off. The staff also states that " actual depressurizations event have not demonstrated significantly increased leakage." This observation is completely irrelevant. Actual depressurization events have occurred under conditions where the tubes were covered with water and therefore the jet could have been quenched and had not sufficient localized energy to penetrate adjacent tubes.

As already discussed under Issue 2, the potential for large accident leakage lowers the probability that the operator will be able to control the accident and therefore the probability of a core melt is controlled by the probability of a steam line break with containment bypass which was estimated in Attachment 4 as 10^{-4} per reactor year. Independent RES analysis in September 1992 predicted LERF of $3.4x10^{-4}$.

3.3.2 Severe Accidents. The staff LERF value of $3.9x10^{-6}$ is based entirely on the analysis of NUREG-1570 which relies on the Westinghouse 1/7th scale mixing tests to show that the surge line will fail before the tubes and therefore degraded tubes will not impact severe accidents. The Westinghouse tests are not applicable to the calculations of tube temperatures because they were conducted without leakage. As discussed in Attachment 3 and finally confirmed in the DPO Consideration Document even relatively small leakages (600 gallons per minute) are of the same order as the free convection loop flow. Steam generator tubes will reach higher temperatures at a shorter time than would have been predicted by the Westinghouse data.

4. Iodine Spiking Issue

4.1 Staff Position

The staff states that since data does not exist to show large increases in iodine spiking during depressurization the factor of 500 is adequate to maintain Part 100 Limits.

4.2 DPO Position

Iodine spiking increases of 10,000 times in coolant activity have occurred, Attachment 8. Iodine spiking occurs when, power, temperature or the pressure are changed. Since MSLB is a large depressurization event the use of 500 is not adequate to assure that Part 100 limits will be maintained.

As stated in Attachment 2, at the October 2, 1997 ACRS meeting I agreed that this issue could be resolved. This was based on the original draft of the DPO consideration document and staff presentations to the ACRS at that time. Since the ACRS meeting, the staff made material changes to the initial DPO Consideration document and therefore the iodine spiking issue remains unresolved.

4.3 Discussion

The DPO Consideration Document reviewed the literature and conducted computer studies to conclude that there was not sufficient data to determine the increase in spiking during MSLB events. For this reason it was decided to do nothing. I agree that this problem is very complex, and for this reason, uncertainties must be incorporated in the 500 iodine spiking factor. Alternatively, uncertainties may be included in the leakage rate to account for spiking. The difficulty in dealing with the iodine spiking issue can be traced to the very low concentrations of iodine in the coolant. Because of this, the mean free path between molecules is too large for classical chemical partitioning coefficients to apply. Such ignorance, does not justify a plant practice of lowering coolant concentration to show that Part 100 limits are met. Attachment 8, clearly shows that there is a large scatter in the spiking data, and at low iodine coolant concentrations considerable amount of the data falls above a spiking factor of 500. Dr. Powers , ACRS, proposed a method of how to approach this problem, but nothing has been done in this regard. The proposition that since there is no data, nothing needs to be done is unacceptable.

5. Severe Accident Issues

The concern is that the steam generator tubes will fail prior to other portions of the reactor coolant (surge line) when degraded tubes are allowed to remain in service.

5.1 Staff Position

After years of denials, the staff finally admits that dry steam generator secondary side events, can lead to steam generator tube failures and subsequent containment bypass.

5.2 DPO Position

Consideration of jet erosion on adjacent tubes and NDE uncertainties, indicate that there is a potential for significant primary-to-secondary leakage during certain severe accidents. Because of such leakage, the inlet plenum will not be perfectly mixed and steam generator tube temperatures could reach hot leg temperatures. Failure of the steam generator tubes before surge line failure has been predicted .The proposed resolution of this issue is unacceptable.

In the original DPO Consideration Document the staff stated that changes to licensing basis "will include the risk associated with severe accidents performance of tubes, including leakage consideration". The Farley inspection waiver, Attachment 5 and the final DPO Consideration Document indicate that staff has no intention of evaluating leakage in a meaningful manner. At the summary of the 445th ACRS meeting, when the letter to the commission was being drafted, I agreed that the Severe Accident Issue would be resolved if the staff would develop meaningful guidelines for calculating accident leakage. The final DPO Consideration document is different than its predecessor which went out for public comment. The final document DPPO Consideration states that "the staff will take changes to risk into account when reviewing new repair criteria or methods" This vague approach is unacceptable. "Staff beliefs"t have been used recently as a basis of approving the Farley Nuclear Plant Unit 1 request for inspection relief, Attachment 5.

5.3 Discussion

5.1 Correction. The staff implies that the DPO initially targeted MSLB events and that only more recently it was extended to severe accidents. This is false, misleading, and self-serving. The effect of degraded tubes on severe accidents was discussed in my September 11, 1992 memo as part of the initial DPV package, Attachment 4A,. This memo concluded, strictly on the basis of thermal hydraulic considerations, that allowing degraded tubes to remain in service would increase the risk for containment bypass during severe accidents. This document was available to the staff, but the results were not included in the risk assessment for GL-95 and were not discussed with the ACRS. Had the staff not ignored the September 1992 memo, it would have been more difficult to issue GL-95. In May 1997 the staff informed the Commission that severe accidents risk may significantly be increased. In spite of these findings and the fact that GL95-05 was intended as an interim measure, GL95-05 is being used on a permanent basis.

5.3.2 Crack Size. The staff admits that it does not have data to demonstrate the behavior of very small through-wall cracks under core damage conditions. Based only on judgement, the staff assumed in the Farley analysis that cracks shorter than 0.25 inch will not cause significant leakage. Previously, in NUREG-1570, the staff argued that cracks smaller than 0.25 inch will not exist and 0.25 inch and higher cracks were eliminated because they were equivalent to a rupture. Later, when the staff found that short cracks at Farley could exist indeed and they "solved" the problem by using "beliefs " as a basis that cracks shorter than 0.25 inch will not erode adjacent tubes. To validate these "beliefs" the staff now proposes to conduct research to show that small cracks, unlike larger cracks, will not erode adjacent tubes.

It should be noted that fundamental scaling laws about jet erosion do not suggest that temperature and relative size of the jet opening affect jet erosion characteristics. Velocity, particle concentration, hardness and shape are the main parameters affecting jet erosion.

5.3.3 Jet Confinment. When it was discovered that GL-95-05 presented a potential problem because severe accidents were not considered before the document was released to the public the staff invented a new theory to still justify the use of GL95-05. This theory states that adjacent tubes will not be damaged by jet erosion because the jet will be deflected by the support plate in a direction parallel to adjacent tubes. This model is a pure fantasy. It ignores the fact that cracks may extend all the way to the edge of the support plate as suggested by the Trojan data and basic chemistry consideration. It would be impossible to prove that the pH in the plate support crevice varies sufficiently to exclude stress corrosion cracking at the edges. Even if the jet was flowing in a parallel direction it could still hit adjacent tubes near the upper bends. The support plate/tube gap will contain substantial amount of deposits, preventing the jet from following a well defined parallel path. It is uncommon in engineering to take credit for a very complex flow phenomena without any supporting analysis, especially when the results may have serious consequences.

5.3.4. Plenum Mixing. The staff focuses on the creep behavior of thermally induced failures of flawed tubes instead on damage from jet erosion and unplugging. Nevertheless, the thermal creep results are non conservative because they were based on the false assumption of perfect flow mixing in the SG inlet plenum and that decay heat does not contribute to temperature increase.

As mentioned above and discussed under Issue 3, leakage will prevent perfect mixing in the plenum and may affect heat transport through out the primary loop. It is incorrect to assume that SG tubes will be exposed lower temperature than the hot. In discussing the effect of small cracks on plenum mixing, the staff claims that cracks smaller than 0.12 will have an insignificant contribution to plenum mixing. This implies that the entire tube bundle will be degraded by only one small crack. According to the table provided by the staff, even 50 pinholes, a minuscule fraction of the available tube area, will prevent complete mixing in the inlet plenum. It would appear that the staff failed to multiply the leak from one pin hole (0.06 lbs/sec) by the number of potential pin holes.

Another incorrect assumption in NUREG 1570 is that decay heat does not affect tube temperature distribution. The Japan Atomic Energy Research Institute,(JAERI) studies show (Attachment 7) that the heat release from fission products causes a sharp rise in steam generator tube temperature, but not in the surge line temperature, proving independently, that steam generator tubes will fail before the surge line does. When the tubes fail first, the LERF exceeds Commission guideline (10E-5 reactors/year). The staff deflected the significance of the JAERI results by claiming, that they are not applicable to degraded tubes because the analysis was based on the unmixed vapor temperature. The JAERI results are correct because as already mentioned above the NUREG-1570 assumption of complete mixing is incorrect.

In conclusion: (1) the staff completely ignored the effect of degraded tubes during severe accidents (2) next, when potential problems with erosion were discovered, the jet erosion phenomena was eliminated by assuming the absence of short cracks, and the presence of jet confinement (3) idealized models which assumed complete mixing and ignored decay heat were used to obtain relatively low temperature in the plenum (4) when short cracks were discovered, the staff resorted to "beliefs" as the justification for allowing Farley relief from inspection (5) research has been initiated on the "effect of crack length on the steam cutting phenomena"

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