ATTACHMENT 6

Regulatory Guide DG-1074 Package Attachment

DIFFERING PROFESSIONAL OPINION REGARDING NRC APPROACH TO STEAM GENERATOR AGING.

U. S. Nuclear Regulatory Commission

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NOTICE: This document was originally prepared in response to staff considerations of the DPO that were attached to draft GL- 98-xx. This week the staff decided to delay the decision whether the GL should be issued. Instead, Draft DG-1074 and the DPO consideration document are being issued to support a technical interchange with the industry. The content of the enclosed document is not affected by the revised staff position.

DIFFERING PROFESSIONAL OPINION REGARDING NRC APPROACH TO STEAM GENERATOR AGING.

SUMMARY

Even though the proposed Generic Letter, (GL) 98-XX has been in preparation for more than 5 years it is not ready yet for public comments. It is too complex, it is difficult to read, and it has no scientific basis. Unlike previous drafts requiring changes to technical specifications, the proposed actions are mostly voluntary in nature. Nevertheless, it is important to realize that staff's conclusions are not supported by science.

The key role that steam generators play in the safety and economics of nuclear plants presents an opportunity to define a risk-informed approach to steam generator repairs, which could serve as a platform regarding other aging components. The GL missed this opportunity; it uses untested models and treats uncertainties entirely inadequately. This memo discusses the technical weakness of GL-98-XX.

By suggesting that licensees may want to change their technical specifications (TS) or submit proposals for alternate repair criteria (ARC) the GL attempts to address several unresolved safety issues that have been documented in a series of Differing Professional Opinion (DPO) documents starting in 1991. To maintain Accident Leakage Integrity, licensees will have to address the DPO- related issues without having the slightest idea what would be acceptable to the staff. The Advisory Committee on Reactor Safety (ACRS) strongly urged the staff to resolve the DPO before implementing new steam

generator guidelines. The staff's latest attempt in resolving the DPO, which is being issued for public comment, is unacceptable to the DPO originator. The ACRS also refused to endorse the staff reply to the DPO.

The staff rationale for the need to change the TS is that stress corrosion cracking (SCC) and not wastage, is the dominant mode of tube failures. Because it is universally accepted that the initiation and propagation of SCC is difficult to predict, this very argument is a reason why it will also be difficult to implement the proposed TS changes without an accepted methodology of predicting crack size at the end of cycle. The GL allows plants to replace the present 40 percent plugging criteria with ARC provided the licensees submit risk assessments with their application. The staff, however, already made such assessments, and even after using non conservative assumptions, found that certain plants will not be able to meet Commission safety goals. This conclusion is in agreement with the DPO estimates that the large early, release frequency (LERF) may exceed 10E-4 (i.e an order of magnitude larger than Commission guidelines).

Aging steam generators in nuclear power plants are a source of radioactive release to the environment. A small leak from any one of thousands of corroded tubes can propagate, from tube to tube, and open a gate for radionuclide escape. Leakage propagation is of concern because it was not considered when the plants were designed.

In response to unexpectedly fast degradation of steam generator tubes, the industry, in the early 90s, proposed a new approach to repairing tubes. The U.S. Nuclear Regulatory Commission(NRC), adapted the industry approach and issued GL-95, which relaxes plugging criteria. Opposing technical views including a formal DPO were ignored. GL-95 was originally intended as an interim measure; now, GL-98 gives it permanent status even though the DPO and the related high priority Generic Safety issue, (GSI)-163, "Multiple Steam Generator Leakage", remain unresolved.

1. BACKGROUND

The Differing Professional View and Differing Professional Opinion (DPV/ DPO) was initiated in December1991 (Ref.1) in a response to staff's approach in dealing with the degradations of the Trojan Steam Generators. Trojan was shut down permanently following steam generator leaks in late 1992. A formal DPO was filed in July 1995 (Ref. 2) when the staff was about to relax plugging criteria, GL-95, without properly addressing the issue of accident induced leakage. The Executive Director for Operation (EDO) responded, (Ref. 3) that the DPO would be resolved as part of the steam generator rulemaking activities. After many delays, the staff proposed a rule methodology (Ref. 4,5). This methodology was discarded when the ACRS refused to provide its endorsement (Ref.6). The documents that were issued at the termination of the rulemaking activities (Refs.4, 5) did not properly address the DPO issues. At the ACRS request (Ref. 6) the staff issued, in October 1997, a more detailed discussion of the DPO which is attached to GL-98-XX. This third attempt to resolve the DPO issue also failed.

An important issued raised early by the DPO (Ref. 7) was the potential increase in severe accident risk from the relaxation of the 40 percent plugging criteria. The staff ignored the DPO and in 1995 relaxed plugging criteria (GL-95) without informing the public during the comment period that there is a potential for risk increase during station blackout accidents. It was only recently (Ref.8), that the staff informed the Commission that severe accidents risk may significantly be increased if alternate repair criteria are used.

The relaxation of the 40 percent plugging criteria for certain types of tube degradations, under GL-95, was intended as an **interim** measure to be incorporated under the new steam generator rule. Since the introduction of the **interim** measures: (a) field experience revealed that the voltage methodology is non-conservative; (b) staff found that alternate plugging rules may increase severe accident risk; (c) the staff was not able to resolve the DPO issues; (d) the staff did not complete the resolution of GSI -163, "Multiple Steam Generator Tube Leakage". **Consequently, several plants are now operating at safety levels below Commission guidelines.**

2. ANALYSIS

2.1 The Problem

The nuclear industry has a large investment in steam generators whose ages vary from several months to more than 25 years. The aging units increasingly develop a variety of cracks that mostly show up as small leaks; these leaks do not pose major safety problems because the affected plant can be shut-down in a timely manner. The real concern arises when plants are subjected to accidents such as main steam line break (MSLB). **Can the leaks become larger during such accidents and propagate from tube to tube?** It is generally agreed that leaks will enlarge during accidents, but there is no agreement how large they can become. The ultimate size of the leak is very important because it determines whether design basis accidents can be brought under control and whether dose releases can meet 10 CFR PART 100 requirements.

Staff's response to the DPO claims that leaks during design basis accidents will not enlarge sufficiently to lead to a core melt. **These differences between the staff and the DPO reside in the calculations of radioactivity leakage through the defective tubes.**

2.2 Staff's Assumptions.

To support their claim that accident induced leakage remains small and can be controlled by the operator, the staff postulated a set of assumptions which are listed below. They neither discuss nor explain the reasons for their selection; decision makers, nevertheless, appear to accept the staffs' final conclusions.

- Leakage is a unique function of voltage as measured by eddy current probes.
- Historical plant data allows for predicting crack growth rates during each cycle.
- Leakage during MSLB accidents depends only on the pressure across the crack.
- The steam generator is filled with water throughout the MSLB accident.
- Support plates will prevent cracks enlargement during SBOs
- Westinghouse 1/7 test data allows mixing predictions during SBO accidents

All the above assumptions minimize the magnitude of accident induced leakage.

2.3 Comments on Staff's Assumptions

A. Why Voltage Measurements Are 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 that penetrate the tube surface. If many cracks are present below the surface of a given tube, the probe will detect a large voltage, but there will be no leakage unless the cracks penetrate the surface. Conversely, a very small voltage may produce a large leakage if a single crack penetrates the surface. In other words, voltage is not a unique function of leakage. The initiation of cracks and their growth is such that one will not expect a correlation between voltage and flow; 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 of samples may exhibit correlation between voltage and leakage. However, because of the non- unique relation between voltage and leakage, the use of the data beyond the laboratory environment introduces uncertainties.

The space between the tube and tube support structure forms a crevice, a flow-starved region, for the accumulation of chemicals that are left behind when the water in the crevice evaporates. Wetting and un-wetting inside the crevice lead to highly concentrated water solutions. Variations in crevice size and tube-to-tube heat transfer cause large variations in crevice-to-crevice chemistry. Since stress corrosion cracking is controlled by the specific chemistry/surface stress of the tube and since it is not possible to measure this parameter, laboratory tests inherently contain large uncertainties that cannot be defined by statistical means. The laboratory-generated database includes some samples of failed tubes that were remove from service. The chemical environment and degree of deformation that these samples have undergone on removal are unknown. When real world environments can not be simulated in the laboratory, it is a common engineering practice to exercise conservatism.

B. Why Historical Statistical Crack Behavior Cannot be Used to Predict Future

Crack Growth.

It is commonly accepted (Ref. 9) 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 are controlled by numerous electrochemical, metallurgical, and stress variables. At a given stress level, crack growth 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, it cannot be expected that crack growth and topography will remain the same for all cycles. Operating experience (Ref.10) clearly demonstrates that relying on the above procedure for predicting crack growth rates is nonconservative. For example, Farley's data show that crack growth rates for cycle 14 were higher than those for the two previous cycles--13.7 volts were measured, significantly above the predicted peak of 7.6 volts. During MSLB accidents, the presence of even one through-the-wall crack can cascade the accident, leading to a core melt.

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 are governed by local stress and chemistry. Even when cracks propagate slowly or are in arrest, other cracks are being initiated and then interact with previous cracks to form a complex network of cracks. Cycle-to-cycle voltage measurement reveals nothing about crack arrest/growth cycles.

Because of considerable industrial experience in stress corrosion the common engineering practice (Ref.9) is not to operate with components susceptible to stress corrosion cracking. By allowing operations through the wall cracks the NRC is setting a new precedent in material sciences.

C. Why Leakage Guidelines are Not Conservative: Leakage During MSLB Accidents Does Not Depend Only on Pressure.

The staff approach to setting limits on accident induced leakage is based on selecting a failure scenario and then showing that the resultant leakage is limited by Emergency Core Cooling (ECC) pump capability (Ref.11). The leakage is assumed to coincide with crack enlargement from pressure loads during the MSLB accident and its magnitude depends only on the number of pre-existing defects, their size, and the maximum pressure during the accident.

The assumption that the flow area is enlarged during the accident by pressure alone leads to low accidental leakage (< 100 gpm). Industrial experience, however, indicates that there are other sources for enlarging cracks beside pressure. Damage from jet erosion and sudden unplugging of cracks during transients are two mechanisms for crack enlargement and leakage increase.

Jet Erosion. During MSLB and SBO accidents leakage can be increased by jet erosion. The jets emerging from pre-existing through- the- wall cracks contain small water droplets and micron size particles which impact the adjacent tubes at velocities exceeding 2500 ft/sec. Abrasive steam jets are known to initiate and propagate large leaks in Kraft boilers, leading sometimes to large steam explosions. Damage to turbine blades is another example where wet steam causes erosion damage. Abrasive water-jet machining and abrasive water-jet pipe cleaning technologies also indicate the potential for damage propagation, during accidents from jet erosion. Using jet-erosion data, the staff agreed (Ref.4) that a steam generator tube can fail in less than 1 minute from water jet erosion.

Effect of Deposits. Cracks are normally filled with salt deposits that may plug and therby limit or prevent leakage under steady operating conditions. Pressure or thermal transients, however, may dislodge these deposits, causing an abrupt change in leakage. The Pacific Northwest Laboratory (PNL) studies (Ref.12), show that sudden leakage changes, through steam generator tube cracks occurs in unpredictable and random manners. The regulatory guide recommends that hydrostatic pressure tests at room temperature be conducted to demonstrate accident-induced leak rate performance. It is a common industrial experience that boiler tubes do not leak under hydrostatic tests, but do leak when the vessel is in operation.

D. Why Steam Generator Will be Dry Following MSLB Accidents, Thereby Exposing Tubes to a Damaging Environment

During the preparation of GL-95 the staff refused to analyze the consequences of jet erosion on risk. After the release of GL-95, the staff analyzed (Ref. 6) the consequences of jet erosion and concluded that jet erosion will rapidly damage adjacent tube during severe accidents, when the secondary side is dry. Unlike during severe accidents, the staff claims that during design basis accidents erosion will not take place because the water in the vessel will attenuate the jet's energy before its impact on adjacent tubes. This assumption appears to contradict the basic laws of physics, because as soon as the pressurized (1000 lbs/sq. in) water in the steam generator is exposed to the atmosphere, the vessel empties; it will take about 30 minutes to refill . Using staff's own data on erosion rates (Ref.6) this time is more than sufficient to propagate leakage throughout the steam generator.

F. Why Support Plates Will Not Prevent Erosion Damage to Adjacent Tubes

For free span cracks, the staff concluded that adjacent tubes could be rapidly damaged. They claim that for cracks in the support plate region, the adjacent tubes will not be damaged because the jet will be deflected by the support plate in a direction parallel to adjacent tubes.

This assumption is based on the belief that the jet will be initiated away from the edges of the plate and will follow the gap in a prescribed manner. This ideal model doe not take into account the fact that the jet can originate at the edges, rapidly enlarging the crack, and impinge on the adjacent tubes. The support plate/tube gap will contain substantial amount of deposits, preventing the jet from following a well-defined parallel path to the tubes.

F. Why Westinghouse-1/7 Tests Should Not be Used to Determine Tube Temperatures During Station Blackout Accidents

To demonstrate that cracked tubes have no impact on the consequences of station blackout accidents, the staff relied on the W - 1/7th scale mixing tests (Ref. 4). These tests are completely irrelevant to the assessment of how defective tube will behave because they were conducted without leakage and without fission product deposits. As discussed in the DPO (Ref. 13) and confirmed by the staff even relatively small leakages (400gpm) are of the same order as the free convection loop flow. Recent analysis by JAERI shows that the heat release from fission products causes a sharp rise in steam generator tube temperature, but not in the surge line temperature. These results add weight to the analysis of Ref. 7 which predicts that steam generator tubes will fail before the surge line does. **When the tubes fail first, the LERF exceeds Commission**

guideline (10 E-5 reactors/year). The staff deflected the significance of the JAERI results by stating, (Ref. 14), that these results are not applicable to degraded tubes because the analysis was based on the unmixed vapor temperature. This illustrates how the staff's practice of neither stating nor discussing important assumptions can mislead decision makers. As already stated above, the staff without providing any justification for neglected leakage, used complete mixing in NUREG-1570. They continue to use this perfect mixing assumption by rejecting the JAERI results.

With degraded tubes the leakage can not be ignored, the unmixed vapor temperature is the proper temperature to use in the analysis of fission product deposits and therefore the JAERI results are indeed applicable. In addition, ACRS consultants also raised concerns about the applicability of the W-1/7 tests and limited range of sensitivity studies. These comments were completely disregarded.

3. DISCUSSION

Failures of steam generator tubes as a result of corrosion can have catastrophic consequences when they occur during steam line break accidents. The underlying reason for severe corrosion in steam generators was the selection of alloy 600 during the design phase of the plant. Replacement steam generators use the less susceptible alloy 690; however, not all the plants have replaced their steam generators yet. Stress, coolant chemistry, material microstructure, and temperature are the major factors that affect corrosion. Since these factors vary from plant to plant, it is critical to perform a risk-informed Fitness-for-Next Cycle Assessment during each inspection interval. Given the observed crack sizes and their locations, conservative K values can be used to estimate the total potential leakage area under MSLB at the end of the next cycle. In selecting K values, one should reference his/her source and discuss how crack linkage is accounted for in the analysis. Once the initial leakage has been calculated, an estimate should then be made to determine leakage propagation during MSLB, taking into account erosion and crack enlargement from unplugging. The final conclusion from this assessment should indicate how many tubes should be plugged and whether a mid-cycle inspection is required. At this point, cost becomes an issue and should be factored into the decision and discussed in the assessment. The analysis does not have to be lengthy or very elaborate, but it should reflect that experienced and qualified engineers with familiarity with the history of the plant have prepared the assessment. Assumptions should be consistent with engineering observations e.g. if the calculated jet velocities for a given crack geometry are low, it would be sufficient to state that erosion has not been observed on material 600 or similar materials at these velocities and therefore the potential risk for tube-to-tube propagation is very low.

As part of their licensing basis, plants are required to meet certain accident induced leakage criteria. Changes to these criteria require that the staff review the consequences of such changes on accident leakage. Stress corrosion cracking introduces a phenomena where it is very difficult to predict leakage because there is hardly any data on flow through these complex cracks. The use of voltage-based repair limits is a convenient tool to bypass these difficulties even though it has no scientific basis. Nevertheless, if this approach is used, uncertainties must be assessed and not ignored. To gain acceptance with the public, performance-based regulation must be transparent. At the least, the public should be informed that because of lack of other means, non-scientific methods are being employed.

The NRC has indicated that it intends to move toward risk-informed approach to make regulation more efficient. Before this can be accomplished the agency must first institute procedures that will protect the public from regulations which are based on unreliable technical information. The supporting documents (Refs.4, 11) clearly demonstrate that present internal controls are not adequate to prevent the use of irrelevant data to advance a given outcome. For example, one is led to believe (Ref.15) that large leakages will not occur because such leakages have never occurred during past depressurization events in operating reactors. One is not told, however, that these depressurization events are not relevant to MSLB accidents, because the plants were not operating with through-the-wall cracks and the steam generators were filled with water.

GL-98-xx states that the risk for plants that operate under the ad- hoc provisions of GL-95 remains at an acceptable level. This statement contradicts the information the staff provided to the Commission (Ref. 8) stating that some plants using alternate repair criteria may not meet Commission guidelines. The response to the DPO, however, states that when the interim plugging criteria were considered for steam generator tubes under NUREG-1477 and GL-95-05, they did not have the benefit of the Probability Risk Assessment (PRA) policy statement, to focus attention on severe accidents. In other words, the staff has not conducted a complete risk assessment before relaxing, and continuing to relax existing plugging rules. The lack of a formal policy does not bar the staff from performing analysis when there are clear signals that a potential problem exists.

The staff claims that the original DPO was limited to Design Basis Accidents (DBA) accidents, only, and did not include severe accidents. This is incorrect. Several years before the preparation of GL-95-05 a DPV/ DPO dated Sept. 11 1992 (Ref. 7) concluded that degraded tubes may have an important effect on severe accident risk. The staff ignored this observation and severe accident risk was not considered. Only after the release of GL-95-05 , did the staff discovered that a significant increase in risk exists during severe accident. This finding still did not prevent the staff from continuing rule relaxations under GL-95-05.

During the development of the GLs and the steam generator rule in the last 6 years, the staff continuously kept changing their actions even-though no new relevant data was being generated. After 4 years of rule-making activities and shortly after briefing the Commission that the new steam generator rule will set a precedent in rule-making, the SG rule was withdrawn with an explanation that this was dictated by cost benefit considerations. After spending 2 years on preparing its successor, GL-98-XX, the GL was withdrawn with the explanation that this was dictated by the NEI-97-06 initiative (Ref. 16). Since the NEI initiative was issued in December 1997, it is difficult to understand why the staff did not reach an agreement with the industry earlier, instead of expending resources on GL-98-XX. NEI-96-06 is a high level document which does not contain difficult issues. While the resolution steam generator issues is being delayed the staff, at the same time, relaxes the existing plugging rules on an ad-hock basis and with disregard of Commission safety guidelines.

4. CONCLUSIONS

GL-98-XX and GL-95-05 demonstrate approaches which are based on adapting models to attain a desired outcome. Such approaches are not conducive to gain endorsements of risk based regulations. Unbiased risk assessments should include the following elements:

1. Good science
2. Timely response to steam generator problems
3. Acceptance criteria for accidental leakage assessment
4. Use of relevant data to support conclusions
5. Proper use of referenced material
6. Sensitivity studies with full range of plausible parameters.
7. Proper use of different views and opinions

5. RECOMMENDATIONS

1. **Do not issue GL-98-XX for public comments**
2. **Phase out GL-95-05**
3. **Issue specific acceptance criteria to demonstrate that accident induced leakage will meet Commission safety guidelines and Part 100 requirements**
4. **Reach agreements with the industry on methods for measuring safety performance of degraded steam generator tubes.**

5. REFERENCES

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