CP&L

Carolina Power & Light Company

SERIAL: NLS-85-266

JUL 18 1985

Mr. Hugh L. Thompson, Jr. Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

COMMENTS ON NUREG-0844,
NRC INTEGRATED PROGRAM FOR THE RESOLUTION
OF UNRESOLVED SAFETY ISSUES A-3, A-4, AND A-5
REGARDING STEAM GENERATOR TUBE INTEGRITY

Dear Mr. Thompson:

Carolina Power & Light Company appreciates the opportunity to comment on NUREG-0844. The staff recommended actions presented in the document are good engineering practices that in most part have been part of the licensees' operating practices. In our judgement, there are no compelling safety issues to warrant imposing generic requirements.

We are concerned that the staff may not share this view. Although Generic Letter 85-02 did not explicitly state that the staff recommended actions should be imposed on licensees, NUREG-0844, Section 1.7 states that "effective programs in these respects are important to safety and indeed are a requirement of the Federal Regulations." Based on this conclusion, the staff could establish new requirements from what now are recommended actions, since the staff is obligated to impose actions which it believes to be necessary in order to comply with regulations.

We disagree with the staff's conclusion, and have prepared detailed comments (enclosed) which we hope the staff takes into consideration before it publishes the final document.

We believe that attempts to make regulatory requirements of good engineering practices is counterproductive to future research efforts. Private industry should be allowed to develop standards and goals which reflect the state of the art without the threat of becoming requirements. Such standards and goals are generally important to the licensees from an economic standpoint. However, the viable degree of implementation of these goals and standards varies from design to design. Therefore, the licensees should have the freedom to select the alternatives that best suit their needs and, if necessary, develop new alternatives. An example of this is the PWR Secondary Water Chemistry Guidelines (SGOG Special Report EPRI-NP-2704), which the NUREG appears to endorse as a requirement. If examples like this one repeatedly become requirements, licensees will be reluctant to develop new alternatives and may oppose other industry groups from doing so. The NRC should adopt policies which encourage new initiatives, not deter them.

411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

A058
1/1

If you have any questions, please contact Mr. Pedro Salas at (919) 836-8015.

Yours very truly,

S. R. Zimmerman
Manager
Nuclear Licensing Section

PS/crs (1683PSA)

Enclosure

cc: Mr. T. Cox

CAROLINA POWER & LIGHT COMPANY

Comments on NUREG-0844
NRC Integrated Program for the Resolution of
Unresolved Safety Issues A-3, A-4, and A-5
Regarding Steam Generator Tube Integrity

NUREG-0844	COMMENT
1 EXECUTIVE SUMMARY 1.6 Disposition of Potential Industry Actions 1.6.1 Staff Recommended Actions In view of the relatively low generic risk estimates associated with steam generator tube rupture events, none of the potential industry actions shown in Table 1 would be expected to provide a significant and demonstrable reduction in risk if they were to be implemented as generic requirements. However, the staff's value impact evaluation indicates that several of these potential industry actions as a group are effective measures for significantly reducing (1) the incidence of tube degradation, (2) the frequency of tube ruptures and the corresponding potential for significant non-core melt releases, and (3) occupational exposures and are consistent with good operating and engineering practice. As a group, these actions are also effective measures for mitigating the consequences of SGTRs. Adoption of these actions by licensees would further reduce public risk . . .	The assertion that "adoption of these actions by licensees would further reduce public risk" does not recogize the fact that most of these actions are currently part of the licensees' operating practices. The staff selected for recommendation eight actions from a list of seventeen based on generic value impact and probabilistic studies. Plant specific · benefit in reducing probability of core melt, · benefit in reducing occupational radiation exposure, · net economic benefit, and · benefit in prevention of significant non-core melt releases, will vary from plant to plant depending on the specific design and the conduct of operation. Therefore, the selection of actions and the degree to which each action is adopted is a management decision that should be left to the organization most familiar with the plant, that is, the licensee's technical and managerial staff. The recommendations are good engineering practices and, as such, should remain that way.

NUREG-0844	COMMENT
1.6.2 Potential Industry Actions Warranting Further Staff Study Apart from potential industry actions which the staff has dispositioned as staff recommended actions, the staff has concluded that others of these actions merit further study by the staff . . .	The staff should re-evaulate the need to further study, from a regulatory standpoint, the actions covered under this section. The recommendations are not essential to protect the health and safety of the public. They are good engineering practices. Therefore, additional research should not be performed by the regulatory agency (NRC) but should be left to the industry (e.g., EPRI and vendors) or the Department of Energy.
1.6.3 Deleted Potential Industry Actions The staff has concluded that the remainder of the potential industry actions identified in Table 1 are not appropriate as generic staff recommended actions, nor do these actions warrant additional study as a staff action . . .	Even though this paragraph states that the actions discussed in this section do not warrant additional study, the recommendation in Section 2.7 (categorized as deleted) contains a caveat which appears to indicate that the staff will continue to spend resources for its study. This is another instance where, considering the staffs conclusions, the research should be left to the industry and not the regulatory body.

CAROLINA POWER & LIGHT COMPANY

Comments on NUREG-0844

NUREG-0844	COMMENT
1.7 <u>Issuance of Generic Letters Regarding Staff Recommended Actions</u>	
The fact that the staff has found public risk from steam generator related causes to be relatively small does not diminish the need for effective programs by licensees to maintain steam generator tube integrity and for mitigation of steam generator tube rupture events. Effective programs in these respects are important to safety and indeed are a requirement of the Federal Regulations previously cited. The Staff acknowledges that the industry has made significant progress in improving steam generator reliability . . .	CP&L disagrees with the staff's conclusion that the staff recommended actions are required by Federal Regulations. This is a far reaching conclusion that should not be buried in the text of the NUREG. If the staff considers this to be the case it should highlight it in the cover letter to the generic letter. The need for an effective program is not a safety concern, but is a good practice concern. The staff can recommend good practices to licensees, however, the staff should not impose them.
1.9 <u>Conclusions Stemming From the Integrated Program</u>	
1. The staff's integrated program has found public risk from SGTR related causes to be small.	1. CP&L agrees that public risk from SGTR related causes is small.

NUREG-0844	COMMENT
1.9 (Cont'd) 2. The Staff's integrated program has reaffirmed the need for effective programs by licensees to maintain steam generator tube integrity and for mitigating steam generator tube rupture events. Effective programs in these respects are important in assuring public health and safety and are a requirement of 10 CFR Part 50, Appendices A, B, and 10 CFR Part 100 . . .	2. CP&L disagrees with the staff conclusion that the recommended actions are required by the regulations. How has the staff "reaffirmed the need for effective programs "when it has" found public risk from SGTR related causes to be small." The two conclusions are not consistent.
1.10 <u>Basis for Continued Plant Operation and Licensing</u> Pending Technical resolution of USIs A-3, A-4, and A-5 with final publication of this report, the NRC staff concludes that continued operation and licensing does not constitute an undue risk to the health and safety of the public.	The staff states that publication of NUREG-0844 will constitute final resolution of USIs A-3, A-4, and A-5, therefore, the additional technical studies that the staff wishes to continue are not warranted (as indicated in comments to Sections 1.6.2 and 1.6.3).

NUREG-0844	COMMENT
2 VALUE-IMPACT EVALUATION OF POTENTIAL INDUSTRY ACTIONS	
2.1 <u>Prevention and Detection of Loose Parts and Foreign Objects</u>	
2.1.1 Secondary Side Visual Inspections & Improved QA Procedures	
Visual inspections of the steam generator secondary side and improved quality assurance/quality control work procedures should be implemented for the prevention of loose parts and foreign objects.	Visual inspections of the steam generators secondary side and improved quality assurance/quality control work procedures are part of current operating practices. Details and extent of the implementation should be left to the licensees.
. . . SAI estimates that the cost required to implement these two actions (about $0.2M per plant) is more than offset by the economic savings (> $3.1M) resulting from implementation . . .	As indirectly acknowledged by the staff, the implementation of these actions is an economical concern and, as such, should be implemented at the discretion of the licensees.
. . . These actions could potentially reduce loose parts related SGTRs by as much as an estimated 90 percent. Since two of the four SGTRs to date have been loose parts related, this translates to a 45 percent reduction in the overall SGTR frequency.	CP&L questions the use of four data points to draw statistical conclusions with regard to the reduction in overall SGTR frequency. It also appears that the staff has assumed in order to draw its conclusions that none of the proposed actions have been in effect previously. This leads to overestimation of the effectiveness of the recommendations.

NUREG-0844	COMMENT
2.1.2 Loose Parts Monitoring System (LPMS)	
. . . Given the implementation of appropriate secondary side visual inspections and improved QA procedures, the staff concludes there is insufficient justification at this time to support implementation of a LPMS system as an additional staff recommended action. However, as discussed in Sections 2.1.1.1 and 2.1.1.3, some utilities may prefer to implement an LPMS in lieu of secondary side visual inspections.	Even though the staff recognizes that use of a LPMS has an effectiveness comparable to the inspections, Generic Letter 85-02, Recommendation 1.a does not retain that thought and could lead to the misinterpretation that inspections should be required inspite of a LPMS.
2.2 Steam Generator Tube Inservice Inspection	
2.2.1 Supplemental Tube Inspections	
. . . The current requirements for inservice inspection frequency and scope are based primarily on experience, engineering judgement, and practicality. The required frequency was based on the frequency of refueling outages so that regular ISI would not unnecessarily impact plant availability and incur needless expense. The required scope of ISIs also was established primarily on the basis of experience and judgement with the goal of achieving safe operation of steam generators by selecting a representative tube sample and minimizing personnel exposure. No analysis has been performed which included:	The staff's comments appear to indicate that there has been a total disregard to consideration of multiple tube ruptures during a design basis accident. This is not the case. Multiple SGTR is a very unlikely event. The staff's own evaluation (NUREG-0844, Section 3) shows very low probabilities even though they were calculated using unrealistically conservative assumptions (as discussed in the comments to Section 3). Therefore, experience, engineering judgement, and practicality are reasonable basis for determining frequency and scope of inservice inspection.

NUREG-0844	COMMENT
2.2.1 (Cont'd)	
(1) a system and accident evaluation to establish the limiting number of defective tubes that can be tolerated to fail during design-basis accidents and (2) statistical determination of the required scope of inspection to ensure that no more than the limiting number of defective tubes will not be inspected . . .	The staff should re-evaluate the need to study further, from a regulatory standpoint, this action.
. . . Based on the results of the staff's value-impact analysis (described in Section 2.2.1.3), the staff has concluded that this potential industry action is not appropriate in its present form for inclusion as a staff recommended action. As discussed in Section 2.2.1.4, the staff will undertake further evaluation of the supplemental tube inspection sampling issue as a staff action.	

NUREG-0844	COMMENT
2.2.2 Full-Length Tube Inspection . . . The Standard Technical Specifications (STS) and Regulatory Guide 1.83, Part C.2.f, currently define a U-tube inspection as meaning an inspection of the steam generator tube from the point of entry on the hot-leg side completely around the U-bend to the top support of the cold-leg side. The staff recommends that tube inspections should include an inspection of the entire length of the tube (tube end to tube end) including the hot leg side, U-bend, and cold leg side . . .	The need and extent of the inspection should be determined on a case by case basis taking into consideration all the plant-specific measures available at the plant for detection of loose parts and degraded tubes. The Technical Specifications requirements should reflect this plant-specific need for inspection. However, the decision to perform additional inspections which may be desirable because of possible economic benefits should be left to the licensee and should not be managed by Technical Specifications.
2.2.3 Denting Inspections . . . Generic implementation of generic denting criteria would not be expected to result in a significant reduction in SGTR frequency, core melt risk, or in the probability of significant non-core melt releases. However, as is discussed further in Section 4.2.2, the availability of generic denting criteria could result in a net cost savings to the NRC in terms of future review effort. Therefore, the staff will untake further study and development of generic denting criteria as a staff action as discussed in Section 4.2.2.	The staff should consider the economic impact of its actions on the taxpayers. Even though the NRC may obtain cost savings in future review efforts, new requirements would translate in a larger burden to the public. Therefore, the action is against the public interest and as such goes against the spirit of the Atomic Energy Act of 1954. The staff should re-evaulate the need to spend additional resources in this area.

NUREG-0844	COMMENT
2.2.4 Steam Generator Inservice Inspection Interval	
The maximum allowable time between eddy current inspections of an individual steam generator should be limited in a manner consistent with Section 4.4.5.3 of the Standard Technical Specifications, and in addition should not extend beyond 72 calendar months . . . . . . The potential reduction in the baseline SGTR frequency, given generic implementation of this recommended action, was not specifically quantified by SAE, but is believed to be small relative to the potential reductions associated with the staff recommended actions for prevention and detection of loose parts and foreign objects (Section 2.1) and improved secondary water chemistry control and condenser inspections (Sections 2.5 and 2.6) . . . . . . A maximum 72-month inspection interval per steam generator reflects accumulated operating experience, is consistant with good engineering judgement regarding the need for periodic inspections as part of an effective program to ensure steam generator tube integrity involves minimal adverse impacts and thus has been incorporated as a staff recommended action.	The plant-specific inspection frequency should be determined on a case by case basis taking into consideration all the plant-specific measures available at the plant for detection of loose parts and degraded tubes. The Technical Specifications requirements should reflect this plant-specific inspection interval. However, the decision to shorten the inspection interval because of possible economic benefits should be left to licensees and should not be managed by Technical Specifications.

NUREG-0844	COMMENT
2.3 Improved Eddy-Current Test (ECT) Techniques	
. . . The SAI analysis indicated that the occupational radiation exposure (ORE) and economic impacts are small and are more than offset by positive ORE and economic benefits . . . . . . improved ECT procedures could potentially have averted one of the four steam generator tube rupture events to date; namely the tube rupture event at the R. E. Ginna plant. Thus, the staff estimates that improved ECT techniques could potentially reduce the probability of rupture by up to 25% if implemented as a stand-alone action . . . [and when implemented with the other staff recommended actions] the additional potential incremental reduction in rupture frequency . . . is about 10% . . .	The decision to adopt measures that may produce positive ORE and economic benefits should be left to the licensees. Such decisions can only be properly evaluated with consideration of the plant specific design and other measures that may be in place. The staff's estimates on the potential reduction of the probability of rupture is based on very limited data, four data points, and even if such sampling set was adequate, the staff should consider what are the real probability that improved ECT techniques over the techniques that were in place at Ginna could have averted the incident.
. . . The staff has concluded that additional considerations of improved ECT techniques as a generic issue is warranted, but as suggested by the SGOG, this effort should be performed in parallel with ongoing ASME Code Committee activities to develop updated eddy-current inspection procedures for incorporation into ASME Boiler and Pressure Vessel Code . . .	The use of statistical analysis to justify new requirements should be based on thorough analysis of all data available and not in terms of potential. The use of the term potential could mislead the party making the decision implementing (or imposing) the requirement.

NUREG-0844	COMMENT
2.4 Upper Inspection Ports A proposal was made that for all PWR applicants, upper inspection ports should be installed before issuance of an operating license so that visual inspection of upper support plates and inner row U-bend tubes could be performed. Operating plants were not included within the scope of this potential industry action based on a consideration of the ORE and economic impacts of installing ports in an operating plant's steam generator . . . . . . Generic installation of upper inspection ports on pre-operational steam generators would not be expected to produce reductions in SGTR frequency, the probabilities of core melt and significant non-core melt releases, or cost. Although implementation of this action could provide ORE benefits of about 100 person-rems in cases where licensees later decide to install upper inspection ports in an operating steam generator, such benefits will likely be limited to a small number of plants. Thus, potential ORE reductions do not appear to be an important generic consideration. For these reasons, the staff concludes that this potential industry action should not be included as one of the generic staff recommended actions.	CP&L agrees with the staff's conclusion that this issue should not be included as a generic staff recommended action. Furthermore, this study provides justification to preclude the staff from requiring upper inspection ports from individual licensees in the future.

CAROLINA POWER & LIGHT COMPANY

Comments on NUREG-0844

NUREG-0844	COMMENT
2.5 Secondary Water Chemistry Program &	
2.6 Condenser Inservice Inspection	
. . . Licensees and applicants should have a secondary water chemistry program (SWCP) to minimize steam generator tube degradation. The specific plant program should incorporate the secondary water chemistry guidelines in SGOG Special Report EPRI-NP-2704, "PWR Secondary Water Chemistry Guidelines," October 1982, and should address measures taken to minimize steam generator corrosion, including materials selection, chemistry limits, and control methods. In addition, the specific plant procedures should include progressively more stringent corrective actions for out-of-specification water chemistry conditions. These corrective actions should include power reductions and shutdowns, as appropriate, when excessively corrosive conditions exist. Specific functional individuals should be identified as having the responsibility/authority to interpret plant water chemistry information and initiate appropriate plant actions to adjust chemistry, as necessary . . . . . . Licensees and applicants should have a condenser inservice inspection program which addresses the following:	A secondary water chemistry program is primarily an economic concern. Licensees have developed programs to protect equipment exposed to secondary water in order to minimize early degradation which would result in the need for expensive repair work. CP&L recognizes the benefits of an effective program, however, does not endorse making such a program a requirement. CP&L believes that NRC efforts to make regulatory requirements of industry initiatives (SGOG Special Report EPRI-NP-2704) will be contraproductive to future research. Private industry should be allowed to develop standards and goals which reflect the state of the art without the threat that they could become regulatory requirements. Such standards and goals are generally important to licensees from an economic standpoint. However, the viable degree of implementation varies from design to design. Therefore, licensees should have the freedom to select the alternatives that best suit their needs and, if necessary, develop new alternatives.

NUREG-0844	COMMENT
1. Procedures to implement a condenser inservice inspection program that will be initiated if condenser leakage is of such a magnitude that a power reduction corrective action is required more than once per three month period; and 2. Identification and location of leakage source(s), either water or air; 3. Methods of repair of leakage; 4. Methodology for determining the cause(s) of leakage; 5. A preventive maintenance program . . . . . . The staff concludes that these potential industry actions should be incorporated as staff recommended actions.	
2.7 Stabilization and Monitoring of Degraded Tubes . . . The staff finds that there is insufficient basis for a staff position that the industry study this issue further to possibly serve as a basis for new regulatory requirements. The staff will continue to monitor industry practice in dealing with the severed tube issue and will take action on either a plant-specific or generic basis should it later be determined to be appropriate.	Based on its conclusions, the staff should re-evaluate the need to spend additional resources in the area.

NUREG-0844	COMMENT
2.8 Primary-to-Secondary Leakage Limits	
. . . All PWRs that have Technical Specifications limits for primary-to-secondary leakage rates which are less restrictive than the STS limits should implement the STS limits . . .	Plant specific primary-to-secondary leakage limits should be determined on a case-by-case basis taking into consideration plant specific measures to minimize the probability of SGTR. The decision to establish smaller leakage rate limits should be left to the licensees and should not be managed by Technical Specifications.
. . . SAI estimates that implementation of the STS limits at plants not currently implementing these limits could potentially reduce the overall PWR (baseline) frequency of SGTRs by 15% if considered as a stand-alone improvement. . . [and when implemented with the other staff recommended actions] the staff estimates that the additional incremental reduction in SGTR frequency . . . to be approximately 5%.	Considering the staff's assumptions (as discussed in previous comments), reduction in SGTRs may be overly conservative.
. . . The staff finds STS leakage limits are an effective means for ensuring that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a design-basis SGTR or a design basis MSLB . . . The staff has incorporated the STS limits on allowable primary to secondary leakage as a staff recommended action.	

NUREG-0844	COMMENT
2.9 Coolant Iodine Activity Limit	
PWRs that have Technical Specification limits and surveillance requirements for coolant iodine activity that are less restrictive than the STS should implement the STS . . . . . . The staff concludes that the potential industry action concerning coolant iodine activity should be incorporated as a staff recommended action.	Plant specific iodine activity limits should be determined on a case-by-case basis taking into consideration plant specific measures to minimize the probability of unacceptable offsite doses in case of a SGTR.
2.10 Reactor Coolant System Pressure Control	
. . . The staff considered a potential requirement for licensees to evaluate further means of optimizing RCS pressure control with the objective of minimizing primary to secondary leakage . . . . . . The issue . . . has been incorporated as an ongoing staff action item . . .	The staff should re-evaluate the need to spend additional resources in this area.

NUREG-0844	COMMENT
2.11 Safety Injection Signal Reset The control logic associated with the safety injection pump suction flow path should be reviewed and modified as necessary, by licensees, to minimize the loss of safety function associated with safety injection reset during an SGTR event. Automatic switchover of safety injection pump suction from the boric acid storage tanks (BAST) to the refueling water storage tanks should be evaluated with respect to whether the switchover should be made on the basis of low BAST level alone without consideration of the condition of the SI signal . . . . . . The staff concludes that the subject actions will constitute an effective approach to ensuring that plants are in compliance with GDC 21, 23, and 35, and ensuring that plants do not have design features that will, absent proper and timely operator actions during an SGTR event, result in damage to the safety injection system . . . the staff has incorporated the subject potential industry action as a staff recommended action.	The staff includes this item as a recommended action even though it concludes that it has a low benefit in reducing probability of core melt, no benefit in reducing occupational radiation exposure, negative economic benefit, and no defined benefit in prevention of significant non-core melt releases. CP&L agrees that the actions may have a positive impact. However, the decision to implement the recommendation should be left to the licensee based on plant specific considerations.

NUREG-0844	COMMENT
2.12 Containment Isolation and Reset A proposal was made that all licensess should review and evaluate the response of the letdown system to containment isolation and reset signals . . . . . . The potential cost savings associated with not having to replace the rupture disk, as a consequence of overfilling the pressure relief tank and bursting the rupture disk, is estimated to be minor in comparison to the cost of implementing any necessary modifications . . . . . . The staff concludes that this potential industry actions should not be incorporated as a staff recommended action.	CP&L agrees with the staff's recommendation.

NUREG-0844	COMMENT
3 SUMMARY OF RISK ANALYSES FOR STEAM GENERATOR TUBE RUPTURE EVENTS	
3.1 Single and Multiple SGTR Probabilities	
3.1.1 Initiating Event Probabilities	
For cases of single tube rupture, an initiating event probability of 2×10^{-2}/RY is used. This is based on actual operating experience through mid-1982 (i.e., four tube ruptures in 240 reactor-years (RY) of operation for domestic Westinghouse plants; sufficient operating experience is not available for Combustion Engineering and Babcock & Wilcox plants to justify a smaller value for those plants) . . .	The staff should not limit its evaluation to data obtained through mid-1982 for a report prepared in mid-1985. The additional data will probably show that the initiating event probability is lower than NUREG-0844 assumes.
. . . The probability of a multiple tube rupture as an "initiating event" was assumed to be 2×10^{-3}/RY. This corresponds to the 50% confidence value for the upper bound point-estimate for an event that has not yet been experienced (i.e., no multiple SGTRs for 353 reactor-years of operation for PWRs through mid-1982). The probability of many tube ruptures occurring simultaneously is extremely remote. For this analysis we have assumed that the probability of 10 or more tube failures is 2×10^{-4}/RY . . .	Additionally, the staff should not disregard the data available for Combustion Engineering and Babcock and Wilcox plants. The report correlates tube failures with operational concerns (e.g., loose parts, water chemistry, and operational limits). Such concerns are not necessarily related to the NSSS manufacturers.

NUREG-0844	COMMENT
3.1.2 Conditional Event Probabilities . . . The staff has estimated the probabilities of single and multiple tube ruptures as a consequence of a design-basis MSLB or LOCA. These estimates are based in part upon consideration of the four domestic plants which experienced ruptures during normal operating conditions. Each of these four plants, in all likelihood, experienced a limited period during which it was vulnerable to a rupture under the more severe conditions of MSLB or LOCA. This period of vulnerability was terminated after the degradation of the steam generator tubing had progresses sufficiently far to cause rupture under the less severe normal operating conditions. In summary, the probability of a plant being vulnerable to ruptures during design-basis MSLB and, therefore, the assumed conditional probability of failure is presented in the following equations: P ($\geq$ 1 SGTR) following an MSLB = 0.034 P ($\overline{2}$ to 10 SGTRs) following an MSLB = 0.014 P (> 10 SGTRs) following an MSLB = 0.003	The staff calculated these conditional events probabilities based on data through mid-1982 (assuming that mid-1972 is a typographical error) for the three units that have experienced steam generator problems. The data should be extended to mid-1985. This will result in a more realistic value. Additionally, the staff should not limit its evaluation to the three units which have suffered steam generator tube rupture problems. Such a selection results in overly conservative values. The staff should include in its evaluation a more representative sample, including plants that have not been vulnerable to a SGTR as a consequence of a MSLB.

CAROLINA POWER & LIGHT COMPANY

Comments on NUREG-0844

NUREG-0844	COMMENT
3.2 SGTR Events Challenging the Reactor Trip and Decay Heat Rer .val Functions	
From NUREG-0460, the conditional probability for failure of reactor trip is estimated to be 3×10^{-5}. (The Salem 1 ATWS events of February 22 and 25, 1983 indicate that the unreliability of the trip system may have been six times higher than this value; however, corrective actions are expected to reduce the probability of failure to scram to close to the estimate of 3×10^{-5}/demand.) . . .	It is not clear which corrective actions the staff is referring to. The staff first issued intermediate-term actions on July 8, 1983 as a result of the Salem event (Generic Letter 83-28) and the Commission issued on June 26, 1984 a new rule, 10 CFR § 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light water cooled nuclear power plants." The staff should specify the reduction that each of there corrective actions provided.
. . . On the basis of a limited survey of plant data, total-loss-of-main-feedwater events are assumed to occur at a frequency of 1/RY. This number is believed to be conservative for purposes of estimating SGTR probabilities for the entire spectrum of ATWS events . . .	NUREG-1000, Section 4.1.1 reports an average of 0.15 total loss of main feedwater events, per reactor year.

NUREG-0844	COMMENT
3.3 SGTR Events Resulting from Loss-of-Coolant Accidents	
The secondary category of SGTR events leading to core melt involve those sequences that include a LOCA and consequential tube failures . . .	The staff should re-evaluate the probability of a LOCA based on the new fracture mechanic studies performed in the last few years by industry and accepted by the staff.
. . . Event — Probability (1) LOCA (intermediate to large) — 10^{-4}/RY (2) Failure of 10 to 20 tubes — 3×10^{-3} (3) ECCS ineffectiveness — 10^{-1} 3×10^{-8}/RY . . .	As indicated in the comments to Section 3.1.2, the conditional probability for failure of 10 to 20 tubes is probably lower than 3×10^{-3}. CP&L agrees with the staff conclusion that the probabilities of this sequence of events is extremely low.
. . . for cases with fewer than 10 tube failures, the tube failures do not significantly affect the results. Therefore, the core-melt probability would be dominated by the unreliability of the ECCS (i.e., 10^{-2}/demand) and the tube failures would be unimportant. Therefore, for large-break LOCAs, the core-melt probability due to concurrent SGTRs and steam binding-induced delay in core reflood is extremely low.	

NUREG-0844	COMMENT
3.4 SGTR Events in Combination with Loss of Secondary System Integrity or Failure To Achieve Steam Generator Isolation	
This category of events includes single and multiple tube ruptures followed by stuck open steam generator safety valves, or main steamline break, or failure of the main steam isolation valves (MSIVs) . . .	The staff should not solely rely in the data used in WASH-1400. It should blend into this data the operational experience to date including those instances in which the steam generator safety valves have been required to operate.
. . . The probability of a stuck-open steam generator safety valve plays an important role in several of the scenarios of interest. A failure probability of 3.0×10^{-2}/demand has been used based on the data base established for NUREG-75/014 . . .	
In approximately 100 tests no valves failed to reclose. Of these tests, 27 were performed with water discharge. Although the water discharge test did show increased valve chatter and valve flutter, there were no failures. The assumed failure rate of 3×10^{-2}/demand is consistent with an upper-bound failure estimated at a 50% confidence level for 27 successful tests without any failures . . .	

NUREG-0844	COMMENT
3.5 Core Melt Sequences	
A summary of all of the sequences contributing to the core melt probability for SGTR events is shown in Table 7. This table presents the event sequences, core melt probabilities, and associated risk estimates. These risk estimates are based upon consideration of potential releases and calculations of the potential health effect consequences of such releases as discussed below. Since the single SGTR events are based on a best estimate of the probability of a single tube rupture and the multiple SGTR events (including SGTRs in multiple SGs) are based on upper bound probabilities, there is an inherent bias in the analysis which may make multiple SGTR events appear to be of more significance, from a risk standpoint, than they actually are.	CP&L agrees that the risk to the public presented in Table 7 "appear to be of a more significance . . . than they actually are."
3.5.1 Determination of Radionuclide Releases	
None of the core melt sequences listed in Table 7 has been subjected to detailed scenario-specific analysis of potential radionuclide releases to the atmosphere. Qualitative considerations have, however, led to judgements about release potentials based upon anologies to sequences and release characterizations used in the Reactor Safety Study (NUREG-75/014) . . .	It would be prudent for the staff to await the conclusion of the ongoing work for estimating "source terms". Considerable progress has been made since the Reactor Safety Study; that knowledge should be integrated into this (NUREG-0844) study.

NUREG-0844	COMMENT
3.5.1 (Cont'd) . . . The staff has examined the event sequences and their associated release categories given in the Reactor Safety Study (NUREG-75/014) and has typed the release characterizations for SGTR events shown in Table 7 on the basis of similarities of sequences with respect to radionuclide release transport, and deposition or decontamination mechanisms.	
3.7 Conclusions The foregoing risk analysis carried out by the staff leads to the following conclusions:	
1. Although there are significant uncertainties inherent in the staff's analyses, the analyses contain a number of conservatisms to minimize the potential for grossly underestimating risk.	1. The assumptions contained in the analyses may be too conservative (see comments to Sections 3.1, 3.2, 3.3, 3.4, and 3.5).

NUREG-0844	COMMENT
3.7 (Cont'd)	
2. The staff's analyses indicate that the core-melt probability from all SGTR-related causes is small, no more than 4.7×10^{-6}/RY as an industry average. This probability is a relatively small fraction (10% or less) of the overall probability of core-melt events from all causes based on probabilistic risk assessments that have been performed for a number of PWRs. The corresponding risk to the public is estimated to be limited to 2.5×10^{-3} latent facilities and 1.1×10^{-5} early fatalities per reactor year from SGTR accidents associated with core melt.	2. The estimated probabilities may be too high, particularly the estimated risk to the public (see comments to Section 3.5).
3. The probability of releasing significant but less than core-melt levels of radioactivity (comparable to WASH-1400 [NUREG-75/014] PWR release categories 8 and 9) from SGTR-related causes is estimated to be 2.2×10^{-4}/RY. The corresponding risk to the public is limited to an estimated range of 3×10^{-7} to 1.3×10^{-5} latent fatalities/RY.	3. The estimated risk to the public may be too high (see comments to Section 3.5.1).

NUREG-0844	COMMENT
3.7 (Cont'd)	
5. On the basis of the staff and SAI evaluations of the risk from SGTR accidents, as discussed above, the staff finds that SGTR events beyond the design basis do not contribute a significant fraction of the early and latent cancer fatality risks associated with other reactor events at a given site. Furthermore, the risk assessment indicates that the increment in risk associated with SGTR events in a small fraction of the accidental and latent cancer fatality risks to which the general public is routinely exposed.	5. CP&L agrees with this conclusion.

NUREG-0844	COMMENT
4 NRC STAFF ACTIONS AND COMPLETED ITEMS **4.1 Introduction** This chapter includes a discussion of those items identified by the staff as warranting further staff action or study. Table 10 lists each of these staff actions, including the staff's plans and status for completing these actions . . . . . . A number of the staff actions involve broad generic issues extending beyond strictly steam generator related issues . . .Completion of these broad generic tasks are considered to be outside the scope of the staff's integrated program to resolve "Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." . . . The remaining staff actions identified in Table 10 involve other issues related to steam generators which the staff finds warrant further study or action by the staff. However, these tasks do not appear to involve issues relating to significant risk to public health and safety. Thus, resolution of USIs A-3, A-4, and A-5 are not contingent on completion of these actions. Schedules for completion of these actions will be commensurate with the priority nature of the work.	Based on the staff's conclusions that "these tasks do not appear to involve issues relating to significant risk to public health and safety," it should re-evaluate the need to allocate additional resources. Investigation should be left to the industry.

NUREG-0844

4.2 Steam Generator Integrity

4.2.1 Steam Generator Tube Sleeves

Task

Guidance governing the design, installation, and inspection of steam generator tube sleeves shall be developed by the NRC. The guidance will be presented in a revision of the Standard Review Plan (NUREG-0800) . . .

. . . Bases

Some utilities faced with the prospect of derating power have elected to replace the degraded steam generators. Such replacement, however, requires an extended outage and involves considerable cost to the utility and its customers. . . This task will be scheduled pending the availability of staff resources, after assigning resources to approved high and medium priority generic issues.

. . . Status

This has been ranked as a low priority task since it is not expected to result in a significant reduction in public risk, but has been classified as a regulatory impact issue based reduced industry and NRC costs (memorandum, February 23, 1983) . . .

COMMENT

The staff should not revise the Standard Review Plan to include changes which have as primary objective economic concerns. NRC guidance documents should be reserved for issues that impact the health and safety of the public. Management of economic concerns should be left to the managerial staff of the licensees. Economic concerns have to be evaluated in context with other economic concerns for the plant. These concerns are highly dependent on the specific plant design, thus, should not be evaluated on a generic basis.

NUREG-0844	COMMENT
4.2.2 Inservice Inspection Program for Denting	
Task	
The NRC staff should propose a denting inspection program for inclusion in the Standard Technical Specifications (STS). The program should include criteria for establishing the scope of the inspections and acceptance criteria (i.e., denting limit based upon tube restriction or strain) . . .	The staff should not consider changes to the Technical Specifications of a nuclear facility when it has concluded that the change will not result in a significant reduction in public risk.
. . . Bases	The Technical Specifications should be reserved for those issues that have a significant effect on safety. Inclusion of issues which do not significantly impact safety detract the significance of the important issues, thus, maybe decreasing the overall safety of the plant.
At present there is no specific mention in Regulatory Guide 1.83 and there are no specific requirements in the Standard Technical Specifications (STS) to inspect tubes for denting . . .	In addition:
. . . Status	1. Acceptable methods for dent inspection should include eddy current methods.
This has been ranked as a low priority task since it is not expected to result in a significant reduction in public risk, but has been categorized as a regulatory impact issue since it would produce a small reduction in risk and would provide a net cost benefit to the industry and to the NRC (memorandum, February 23, 1983) . . .	2. Less precise inspection methods (normal ELT) should be permissible up to an established "threshold" dent size. More precise methods should be used to measure in excess of the threshold size.

NUREG-0844	COMMENT
4.2.4 Category C-2 Inservice Inspection Requirements	
Task	
The NRC staff should investigate more practical alternatives to the originally proposed potential requirement concern "Supplemental Tube Inspections" which is discussed and evaluated in Section 2.2.1.	The staff should not consider changes to the Technical Specifications of a nuclear facility when it has concluded that the change will not result in a significant reduction in public risk.
. . . For reasons discussed in Section 2.2.1.3, this task is not expected to result in significant reductions in risk. However, effective steam generator tube inspection programs are in important element of an effective overall program to ensure steam generator tube integrity . . .	The Technical Specifications should be reserved for those issues that have significant effect on safety. Inclusion of issues which do not significantly impact safety detract the significance of the important issues, thus, possibly decreasing the overall safety of the plant.
Completion of this task will be scheduled pending the availability of staff resources, after assigning resources to approved high and medium priority generic issues.	In addition, the existing sampling requirements are poorly structured in they do not allow for the application of engineering judgement. 1. C-1 requires an initial sample of minimum size to include all unplugged degraded tubes. However, the sample size is not increased to account for the number of degraded tubes included. Therefore, the more degraded tubes inspected for "new" degradation, the less previously nondegraded tubes are inspected for "new" degradation. C-1 should consider known degraded tubes plus a minimum sample of nondegraded tubes.

NUREG-0844	COMMENT
4.2.4 (Cont'd)	2. C-2 should be based on the result of the C-1 sample of previously nondegraded tubes. Additional inspections should be required in the regions of the steam generator where degradation was found. Tubing in the affected regions should be inspected. 3. With proper selection of C-1 samples and complete inspections of regions with active corrosion, no additional inspections are warranted (i.e., C-3 100 % inspection are unnecessary).

NUREG-0844	COMMENT
4.5 Radiological Consequences	
This section contains recommendations intended to reduce the potential radiological consequences of a steam generator tube rupture and to improve the ability to accurately measure the amount of radioactivity released from the plant.	The potential radiological consequences calulated by the staff may be overestimated (see comments to Section 3.5.1).
4.5.1 Reassessment of Radiological Consequences Following a Postulated SGTR Event	
Task	
The NRC staff should reassess SGTR accidents to determine the effects of releases made for periods substantially longer and through release points other than those previously analyzed. These analyses should specifically address the assumptions in Standard Review Plan (SRP) Section 15.6.3 (NUREG-0800) and address the costs and benefits of requiring revised analyses by licensees.	The re-assessment should include the latest "source term" methodology (see comments to Section 3.5.1).

NUREG-0844	COMMENT
4.5.2 Reevaluation of Design-Basis SGTR	
Task	
The NRC should consider, in conjunction with the tasks identified in Sections 4.3.1 and 4.5.1 of this report, the necessity of reclassifying or redefining the design-basis SGTR . . .	Based on the staff's conclusion that "this task is not expected to result in a significant reduction in public risk," the staff should re-evaluate the need to reclassify or redefine the design-basis SGTR.
. . . Bases	
The general basis for this recommendation is derived from the number of SGTRs that have occurred and the potential existing for SGTR doses exceeding the guidelines of 10 CFR Part 100. However, these doses would occur only if there were an unlikely, but not impossible, set of circumstances as discussed in detail in Section 8.1 of NUREG-0916. In any event it is considered prudent to reconsider the SGTR event and the SRP assumptions and criteria . . .	
. . . Status	
This task is not expected to result in a significant reduction in public risk, and therefore has been ranked as a low priority, licensing issue (memorandum, February 27, 1983)	