May 21, 1999

MEMORANDUM TO: Stuart A. Richards, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Richard J. Barrett, Chief
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: REVISED PROBABILISTIC SAFETY ASSESSMENT BRANCH INPUT TO SAFETY EVALUATION REPORT ON THE CHANGE TO TECHNICAL SPECIFICATIONS AT CALLAWAY PLANT TO ALLOW USE OF FRAMATOME ELECTROSLEEVE STEAM GENERATOR TUBE REPAIR METHOD

On May 12, 1999, the Probabilistic Safety Assessment Branch (SPSB) issued a safety evaluation report (SER) on the proposal from Union Electric Company (the licensee) to allow the use of Electrosleeving for repair of flawed steam generator tubes. The purpose of this revised SER is to address information submitted by the licensee in a letter from Garry L. Randolph, dated May 17, 1999. This information was available to the staff during its conduct of the original review, but was not submitted formally to the NRC and was not addressed in the SER. This revision also provides further clarification of the basis for conclusions drawn in the original SER.

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

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Revised
Probabilistic Safety Assessment Branch
Input to Safety Evaluation Report for
License Amendment for the Callaway Plant to
Use Electrosleeves for Repair of Steam Generator Tubes

This evaluation addresses the Union Electric Company (the licensee) request to change the technical specifications (TSs) at the Callaway plant to allow for the use of a new technology for repairing degraded steam generator (SG) tubes. The method is called Electrosleeve, a structural nickel plating applied to the inside of a degraded tube to form a tube sleeve. The request was submitted by letters from the licensee dated April 12 and September 24, 1996, February 5, June 9, August 8, and September 10, 1997, and February 24, 1998.

The requested amendment seeks to credit the Electrosleeve as the new reactor coolant system (RCS) pressure boundary, without taking any credit for the flawed tube it repairs. Thus, the amendment essentially seeks approval to use a different and previously unreviewed material as part of the reactor coolant system (RCS) pressure boundary. The intent is to install sleeves that could remain in-service for the remaining life of the SGs.

The Division of Engineering (DE) has reviewed Electrosleeves and found that, when installed, they would meet all requirements associated with the design basis of the Callaway plant. However information supplied by FTI indicates that, under the conditions associated with core damage accidents, the Electrosleeve material is substantially inferior to the Inconel alloys used for the tubes and for all currently approved tube repair methods. Evaluated without credit for the flawed Inconel tube, as proposed by the licensee, use of Electrosleeves could produce increments to the large early release frequency(LERF) that are beyond the acceptance guideline of Regulatory Guide 1.174 and would not provide adequate defense-in-depth against core damage accidents.

DSSA staff have performed an assessment of the severe accident risk associated with this amendment. An estimate of the frequency of core damage accidents to which these sleeves are vulnerable (high-dry sequences) has been made. In addition, calculations of the expected thermal-hydraulic conditions have been performed using the best available codes. Finally, experiments and analyses of the sleeve performance has been performed, giving credit for the strength of the parent tube.

In performing this evaluation, DSSA has met with the licensee and evaluated available information to determine if the frequency of sequences challenging to Electrosleeves could be shown to be less than one in one million reactor years. In addition, DSSA has reviewed the Electrosleeve test data developed by Argonne National Laboratory for the Office of Nuclear Regulatory Research to determine if the test results could be used to establish that Electrosleeves are unlikely to fail during severe accidents. The staff has followed all available Commission guidance on the use of PRA and severe accident information in making regulatory decisions.

STAFF ASSESSMENT OF RISK ASSOCIATED WITH SEVERE ACCIDENTS

Material property data supplied by FTI indicates that the Electrosleeve nano-crystalline nickel is subject to rapid grain growth resulting in a substantial loss of strength at temperatures above the normal operating temperature range for steam generator tubing. Because many of the accident sequences included in current probabilistic risk assessments have identified high temperature challenges to steam generator tube integrity during the development of core damage, the staff considered the performance of Electrosleeves under the conditions identified in those studies.

The high temperature challenge to steam generator tubing was originally identified in the Reactor Risk Rebaselining Study documented in NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (1990). The issue with respect to flawed steam generator tubes was explored in detail in NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture" (1998). The latter study indicated that the risk to the public from these severe accidents has been adequately but indirectly controlled by license requirements that are intended to address the accidents that are within the design-basis of the nuclear power reactors. For steam generator tubes, the strength margins specified in the ASME Code for operation and accidents at normal temperatures have been shown to provide approximately the structural capability needed to survive the high temperature challenges associated with severe accidents.

In practice, licensees have been inspecting steam generator tubes for degradation during plant outages. Tubes which were found to have flaws that could not be shown to meet the applicable requirements have either been repaired with sleeves or plugged to remove them from service. This process has been implemented in a conservative manner such that, in the event of a severe accident, there is a low probability that tubing with a flaw that cannot survive the severe accident, high-temperature conditions would be in service at the time. Because all currently approved sleeving and plugging methods employ the same or similar Inconel alloys as the original tubes, tubes repaired with these methods are not expected to contribute to the probability of accident-induced tube rupture. However, Electrosleeves utilize a material that can become substantially weaker than Inconel at severe accident temperatures. Thus, flaws repaired with Electrosleeves cannot be presumed to survive severe accidents without conducting appropriate analyses.

If a significant fraction of the flaws repaired using the Electrosleeving process cannot survive a severe accident, then the application of this repair method in an operating plant could create a population of tubes in service that would fail with a conditional probability close to one in the event that a severe accident occurs. Failure of one or more steam generator tubes during a severe accident creates a direct path for the large amounts of radioactive material released from the damaged reactor core to reach the environment (i.e., a containment bypass sequence). The radioactive material released under these conditions would have substantial public health consequences. As part of the original plan to address risk issues in support of the proposed steam generator tube integrity rulemaking, RES performed offsite consequence analysis using the MACCS code to address the risk impact of steam generator leakage and single and multiple tube ruptures. This work, summarized in a letter to R. Jones from C. Ader, dated August 30, 1996, indicated an offsite population dose of approximately 1 million rem, with no early fatalities. This would qualify as a large early release. The MACCS calculations used

the Surry site parameters and the NUREG-1150 protective action assumptions. Offsite consequences were calculated out to 50 miles.

Commission policy, as provided in its PRA Policy Statement, Commission guidance on "Safety and Compliance" (Yellow Announcement No.114 and COMSAJ-97-008) and other policy guidance, is that it is the staff's responsibility to consider the change in risk as well as compliance with the agency's regulations and other requirements when reviewing an application for a license or modification. When an application meets all established design basis requirements, there is a presumption of reasonable assurance that public health and safety is adequately protected unless the staff (or licensee) identifies a specific risk-significant issue. In the case that an identified risk-significant issue is related to accidents that are beyond the design basis, the staff may still request information from the licensee on the issue, and may decline to approve the application if adequate protection of public health and safety cannot be reasonably assured because of the issue. However, the applicant may decline to accept the burden of demonstrating reasonable assurance of adequate protection with respect to an issue when that issue involves accident conditions that are beyond the design basis of the plant. If so, in order to disapprove the application, it is the staff's burden to demonstrate that the risk issue is of sufficient magnitude to prevent reasonable assurance of adequate protection. In such cases, the staff will conduct analyses and/or experiments as necessary to establish the magnitude of the risk associated with the identified issue.

Severe Accident Sequences Applicable to Callaway

Analyses completed in NUREG-1570 identified classes of severe accidents that have the potential of leading to a thermally-induced rupture of a steam generator tube. In general, accidents that could challenge tube structural integrity are those for which the pressure of the gases in the reactor coolant system (RCS) during a core melt sequence are high, the heat removal capability of the steam generators is unavailable (dry secondary system), and the steam side of at least one steam generator has substantially depressurized. The extended exposure of Electrosleeves to the temperatures calculated for such transients will degrade the pressure retaining capability of the repaired tubes.

The licensee's May 17 submittal provides the basis for their conclusion that the frequency of Callaway core damage sequences that may precede induced steam generator tube ruptures is very small (1.7E-6/r.y).

The scenarios of interest are those with core melt in-vessel with the RCS at pressure, and the SGs dry and depressurized (high-dry). The key characteristics of sequences that are readily obtainable from PRA models are loss of all feedwater and loss of alternate means of core heat removal, e.g., feed and bleed, or loss of inventory. What will drive this frequency higher is a mechanism that fails or seriously degrades both functions. SBO sequences provide one such mechanism, with loss of service water being another way. At Callaway, SBO leaves only the turbine driven AFW pump as a means of providing secondary side cooling and fails feed and bleed, while loss of service water similarly fails the MFW and the motor driven AFW trains, leaving only the turbine driven train, and will eventually lead to a loss of injection systems (i.e. the feed of the feed and bleed) due to loss of CCW. This is somewhat delayed with respect to the SBO case by the thermal capacity of the CCW system.

Since IPEs do not normally identify high-dry sequences per se, the sequences that contribute must be inferred by an interpretation of the event tree sequences. A spot check of some PWR IPEs yielded a typical CDF in the range of 8E-6/r.y for internal events. There are no significant plant characteristics that would lead to the conclusion that Callaway has a core damage frequency much below the PWR average. External event initiators such as fire and seismic, which can be significant contributors to station blackout type sequences, are not included in these estimates. While it is difficult to estimate the additional contribution from external events, more modern plants like Callaway would tend to have lower values as a result of good train separation.

In a submittal dated March 18, 1999, Callaway outlined their probabilistic model of the high-dry sequences. Subsequently, two additional sequences were identified by NRC staff that, depending on the RCP seal LOCA model used, could also result in high-dry conditions. This increased the frequency of high-dry from 4.6E-06 to 6.5E-06/r.y. This number was refined by the licensee, who identified conservatisms in the model. These conservatisms were outlined in the May 17 submittal. For example, it was assumed in the most likely SBO sequence that core damage would occur in 1 hour if offsite power is not recovered. The licensee pointed out that a more realistic estimate of 2 hours to core damage would allow more time for recovery of power, and thereby lower the CDF contribution. Other conservatisms include cutsets which are precluded by administrative procedures, dependencies which would probably not be relevant to the scenario of interest, and insufficient credit for operator action to recover function. Based on more realistic assumptions, the licensee estimated a high-dry CDF of approximately 1.7E-6/r.y.

The staff's position however, is that, while the conservatisms identified were valid, there are also potential non-conservatisms that have not been addressed. One example of a non-conservatism is the estimate of the running failure rate of the turbine driven AFW pump as 1E-4/hour, which is low compared to the values obtained in the recent AEOD study of AFW reliability. As mentioned, the licensee's estimates do not include contributions from external events, such as fires and seismic events. Therefore, while the frequency is likely to be in the low to mid E-6/r.y range, it is not possible to be more definitive without a detailed review of the PRA model.

For the largest single contributor, the early SBO sequence, which corresponds to 1E-06/r.y of the licensees estimate of 1.7E-06/r.y, the principal recovery is restoration of offsite power or a diesel. The former has been incorporated in the model, and the latter, for the short time scale of the accident is not significant. Exercising the SAG-1 to use the diesel driven fire water pump to fill the steam generators, one at a time, would require depressurizing each SG in turn, and would therefore guarantee the high-dry and depressurized condition.

Based on these considerations, and the need to consider external events, the staff concludes that the high-dry CDF is in the low to mid E-6/r.y range, but not as low as the licensee's estimate.

An important question related to the probability of tube failure is the likelihood that the steam generator secondary side is depressurized. The dominant sequences, by the licensee's estimation are the early SBO sequence, and the long term SBO, in both of which the operators would be following procedure ECA-0.0. In ECA-0.0, the loss of all AC power procedure, step 3 instructs isolating of the RCS. Steps 4 and 5 instruct to restore the TD AFW pump and AC

power respectively. The assumption, based on feedback from Callaway staff, is that if this is not achievable, they carry on with the remaining steps in the procedure. Step 10 instructs to isolate the SGs. Step 16 addresses depressurizing the SGs. However, the caution says if the SG level drops to below 4% (35% for adverse containment) depressurization should be stopped. This will occur immediately on a loss of offsite power. Therefore, we would expect the SGs to be dry and pressurized initially. However, in step 23, when core exit thermocouples exceed 1200 deg F SAG-1 is entered. As discussed above this would put the SG in a vulnerable condition.

For the sequences that result from loss of all feedwater, procedure FR-H.1 would direct the operators to depressurize the secondary to allow injection of fire-water. This should occur long before the onset of core damage and would prevent core damage.

An additional potential depressurization mechanism is MSIV leakage. MSIVs are not routinely leak tested for PWRs. Based on some critical flow calculations, and some measurements taken at Indian Point 2, our best estimate is that it would take about 6 hours for a steam generator to depressurize to 600 psia, and 21 hours to reach 100 psia through a leaking MSIV.

As a result of these considerations, the staff concludes that a small fraction of high-dry sequences will result in all four steam generator secondaries pressurized.

Thermal hydraulic analyses indicate that the primary side depressurization to the vicinity of the accumulator setpoint would result in survival of the tubes. The licensee has not addressed this strategy in any detail. However, the staff notes that severe accident guideline 2 (SAG-2) addresses depressurizing the primary. However, as a procedure it is neither clear nor crisp, but rather involves the weighing of pros and cons, and could not therefore be relied upon. Furthermore, it will not help for the late SBO sequences that result from battery depletion, since there will be no dc power to operate the PORVs.

As a result of these considerations, and the sequence analyses performed in NUREG-1570, the staff concludes that a small fraction of sequences will result in a depressurized primary system.

The staff concluded that the frequency of high-dry events was sufficiently high (low-to-mid-10^{-6}/r.y) such that the contribution to risk due to Electrosleeving could not be ignored.

Consequently, the staff analyzed the performance of the Electrosleeves under severe accident conditions.

<u>Evaluation of Electrosleeves Under Severe Accident Conditions</u>

Review of the severe accident sequences identified as important for the Surry plant in NUREG-1570 were compared to the high-dry sequence frequencies found in the review of the Callaway IPE to select conditions for the study of Electrosleeve performance.

Time-dependent temperature and pressure calculations for severe accident sequences were performed to determine the expected failure time of Electrosleeved tube flaws relative to other major components of the RCS. The computer model predictions of the temperatures experienced by the steam generator tubes and the timing of the surge line failure were performed by the Office

of Nuclear Regulatory Research using the best available thermal-hydraulics codes and methods in order to make these estimates as realistic as is currently possible. Such calculations have substantial uncertainties. In addition, the severe accident conditions used in the tests and analyses were calculated for a different Westinghouse design, which is believed to behave in a manner similar to Callaway. The uncertainty in the thermal-hydraulic conditions, combined with the sparse data on Electrosleeve behavior, make prediction of Electrosleeve/tube success difficult. Nevertheless, given the best estimates of uncertainties, technical conclusions related to Electrosleeve performance are justified.

The basis for this modelling is the research conducted in the 1980s at the Westinghouse 1/7 scale test facility as part of a cosponsored EPRI/NRC program to measure natural circulation flow patterns during a severe accident. In addition to the review of testing and modelling issues at the time of the original code modifications, more recently, as part of the NUREG-1570 activity, the staff sponsored an independent review of SCDAP/RELAP5 capabilities for predicting steam generator tube conditions for the purpose of predicting creep rupture of the tubes. This review concluded that the code and the supporting experimental data were adequate for calculating the relative timing and failure of RCS components for the purpose of assessing risk from thermally induced SG tube ruptures. As part of our effort to evaluate SG tube performance during severe accidents in support of proposed rulemaking (NUREG-1570) RES also performed numerous sensitivity calculations addressing uncertainty in heat transfer. Further, as a result of the independent peer review of the SCDAP/RELAP modeling of these accidents we also received comments on analyses to address uncertainties. Based on this work RES recommended that an uncertainty of ±20 K be attached to the calculated tube temperature at the time of system failure (i.e., surge line rupture). In addition to the sensitivity calculations on heat transfer (described in the letter dated December 2, 1996); RES also performed analyses addressing potential synergistic effects associated with simultaneous variations in natural circulation parameters. It is these natural circulation parameters (i.e., mixing fractions, recirculation ratios, and the fraction of SG tubes carrying forward flow) which to a large degree represent the uncertainty in mixing of flow in the steam generator.

For the high primary system pressure sequence the temperature of the faulted loop steam generator tube at the time of surge line rupture is about 960 K with an uncertainty of 20 K. This does not mean that we believe in the certainty of the absolute temperature calculated for the tubes at a specific point in time but that for a given sequence, relative to the calculation of other piping components' temperatures, tube temperatures in this range are appropriate. General consideration of severe accident uncertainty would cause system temperatures to vary but these effects should cause similar changes in both RCS piping (surge line and hot leg) and SG tube temperatures. A more conservative treatment of uncertainty results in an uncertainty range of approximately 50K. Further confirmation or refinement of uncertainty for the electrosleeve review estimate must await additional evaluation based on analysis and/or testing.

The licensee's May 17 letter includes the results of calculations with the MAAP computer code of the severe accident temperatures experienced by hot leg tubes. For the two thermal-hydraulic cases of interest, the maximum calculated temperature is approximately 460°C (as read from a graph). This is well below the failure temperature of 3-inch Electrosleeved tubes observed in tests conducted for NRC by the Argonne National Laboratory. Even with allowance for uncertainty in the thermal-hydraulic calculations, this result would lead to the conclusion that Electrosleeving is acceptable without a limitation on crack length.

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However, we understand that the licensee's calculations were performed with MAAP 3B which was reviewed by the NRC in 1993. At that time, the NRC indicated to NUMARC and EPRI its concerns with the use of the MAAP 3B code for natural circulation phenomena during severe accidents. In general, MAAP 3B is known to produce non-conservative results for cases such as the ones provided by Callaway. Later versions of the code (MAAP 4.0) were developed and compared to the Westinghouse 1/7 scale tests to address the natural concerns. Therefore, the staff has not based its findings on the licensees' calculations using the MAAP 3B Code.

The most important sequence assumed a station blackout accident where the RCS has remained at its safety or relief valve pressure setpoint and the steam generators' secondary side inventories have dried out, and at least one steam generator loses steam pressure. Eventually, the reactor core overheats to the point where the fuel cladding material begins to react with the remaining steam in the RCS, and the temperatures of the reactor core and the RCS piping and components, including the steam generator tubes, begin to rise rapidly. Within tens of minutes of the oxidation process onset, some part of the RCS pressure boundary is expected to fail because of the high temperatures.

Unflawed Inconel tubes have been tested to failure under simulated severe accident conditions by the Argonne National Laboratory. Based on comparison of those results with the expected severe accident conditions described above, the staff concludes that the tubes would survive with a margin many times the variation in the tube failure temperature, even when the uncertainty in thermal-hydraulic calculations is accounted for.

The surge line is predicted to fail 18 minutes before unflawed Inconel tubes would be predicted to fail (if the surge line failure had not relieved the stress on the tubes). Predictions for the probability of earlier failure of flawed steam generator tubes were performed in NUREG-1570, and showed that the risk of containment bypass is expected to be low due to the low probability of a sufficiently large flaw being present in a steam generator tube.

The staff initially evaluated the performance of Electrosleeves without consideration of the flawed Inconel tube, because the licensee requested that the tube not be credited to avoid future tube inspection difficulties (described in section 4.5). The predicted response based on available data and models developed by Argonne National Laboratory shows failure at temperatures well below those expected in severe accidents. Thus, Electrosleeves not backed by a (flawed) Inconel tube would fail before the surge line. Consequently, if Electrosleeves become common in steam generators, this analysis indicates that they would assure containment bypass in the event of the high-dry sequences analyzed. This would be a substantial increase in LERF and loss of defense-in-depth.

When Electrosleeves alone were found to provide inadequate structural integrity to outlast the surge line under severe accident conditions, the staff and Framatome extended the evaluation to include credit for the flawed Inconel tube. Evaluation of the flawed tube/Electrosleeve composite structure is difficult. A predictive model was developed at Argonne National Laboratory (ANL), and physical testing of Electrosleeved tubes with precisely machined axial slits was conducted

independently by Framatome and ANL. Results of the physical testing did not verify the initial model, so conclusions herein are based on the available test data.

The data indicate that through-wall axial flaws 2 inches or greater in length are unlikely to survive severe accident conditions until the surge line fails. If Electrosleeved tube flaws of such lengths become common in steam generators, and the cracks in the Inconel tube continue to propagate through the Inconel during subsequent service, it would create a high probability that at least one of the Electrosleeved flaws would be present and fail in the event of a high-dry type severe accident challenge.

Test data for 1-inch long, through-wall notches indicated survival of the Electrosleeved tube for about 3 minutes beyond the predicted time of surge line failure. Data for ½-inch notches indicates survival for about 10 minutes longer than the surge line. Due to the uncertainties in the RCS component temperatures and failure times in the severe accident computer model simulations, the failure time data (i.e., the margin of 3 minutes) for the 1-inch notches are still not indicative of certain success. However, the physical tests of the Electrosleeves are considered to predict conservatively the behavior of crack type flaws for several reasons. For example, the notches are through-wall for their entire length, which is unrealistic for cracks. Also, the notches start with a several mil wide opening, while cracks are initially tight. Considering these factors, the staff concludes that a 1-inch limitation on the crack length is adequate to assure a low probability of failure of any Electrosleeved free-span, axial crack under severe accident conditions, even when there are a large number of Electrosleeved flaws in a steam generator.

Risk Insights

Although the Commission has directed the staff to consider risk associated with severe accidents when conducting its regulatory functions, guidance is not yet complete in that area when the licensee/applicant has chosen not to address severe accident issues. For cases where the applicant has agreed to address severe accident issues, guidance has been promulgated in Regulatory Guide 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Therefore, the approach and safety principles delineated in that regulatory guide (though not the specific acceptance guidelines) were considered in evaluating the acceptability of the unrestricted use of Electrosleeves. These factors are 1) small increase in risk, 2) maintenance of defense-in-depth, 3) maintenance of safety margins and 4) monitoring of performance.

With respect to risk increase, staff evaluation indicates that Electrosleeves are not expected to increase the core damage frequency at Callaway, but could significantly increase the large early release frequency (LERF), if applied to large, free-span, through-wall, axial cracks. Limited staff review of the Callaway IPE and additional information provided by the licensee indicate a frequency in the mid-to-low 10^{-6}/r.y range for high-dry type severe accidents that potentially increase LERF due to Electrosleeve failure. This range of values is several times higher than the value that RG 1.174 indicates should normally be considered for LERF increases (1×10^{-6}/r.y).

With respect to defense-in-depth, it is important to note that the steam generator tube boundary constitutes two of the three physical barriers provided in the nuclear plant design to protect the public from the radioactive materials in the reactor core. The staff believes that it is reasonable to assume that core damage accidents are intended to be covered by the containment function, even

though some specific phenomena associated with severe accidents were not understood when the design basis was established. The containment function is required in all NRC-licensed plants for the purpose of limiting offsite doses in the event of an accident. Acceptable performance of the containment function is demonstrated by analysis of design basis accidents (DBA) which do not contain all of the physical aspects of the severe accident sequences now modeled in PRAs. However, the DBAs for containment were clearly intended to be a surrogate for such accidents as they were understood at the time. For example, the calculation used to show compliance with siting regulations (10CFR100) are done with a source term indicative of a severe accident. Acceptable containment leakage rates are set using that source term as well. The containment function is a defense-in-depth feature, and is meant to cover the cases where the systems designed to prevent core damage have failed to operate. Unrestricted use of Electrosleeves could negate the containment function and the defense-in-depth principal for an important segment of the core damage frequency.

With respect to maintenance of margins, use of Electrosleeves on large, free-span, axial flaws has been shown to provide little or no margin for severe accidents. This is in sharp contrast to currently approved sleeving methods that employ Inconel sleeves, which are expected to perform similarly to the original, undegraded Inconel tubes.

Finally, with respect to monitoring performance, the applicant/licensee has declined to modify the amendment request to address continued degradation of the Inconel tube where Electrosleeves have been applied. Although there are some technological hurdles to be overcome to make such monitoring feasible, these same inspection difficulties must be successfully addressed in order to effectively monitor the structural integrity of the Electrosleeve, itself. Without such a capability, this amendment would not be in conformance with this safety principle.

In summary, the current application (i.e., to use Electrosleeves without further regard to the condition of the Inconel tube) affects a plant barrier that is important to public protection from severe accidents, can substantially degrade the performance of that barrier in comparison to the currently available and approved technologies, and would not be acceptable if evaluated against criteria for voluntary risk informed applications.

Conclusions and Risk-Related Requirements

On the basis of its assessment of Electrosleeve performance under severe accidents conditions, the staff concludes that it is necessary to restrict the length of Electrosleeved free-span axial cracks to 1" through-wall or equivalent. In addition, if Electrosleeves are to be left in service indefinitely, it will be necessary to develop an inspection method capable of detecting subsequent growth of the crack in the Inconel tube beyond that length limit. However, based on evaluations of crack growth rates by DE, DSSA believes that application of Electrosleeves to cracks initially found to be less than 1-inch in length will not produce an unacceptable increase in risk or decrease in defense-in-depth for the 3 reactor-year period provided by this limited license amendment.

The length limitation on axial cracks to which Electrosleeves may be applied is not expected to impede the intended use at Callaway. Experience from previous in-service inspections at Callaway indicates that the primary mode of degradation likely to be repaired in the near future will be cracks in the expansion-transition region at the top of the tube sheet. Cracking in this area is primarily driven by high residual stresses due to the expansion process. Transition regions for

hydraulic expansions are on the order of 0.5 inches, and, by design, they begin about 0.25 inch below the tubesheet face. Therefore, these cracks are expected to terminate with free-span lengths of 0.25-inch or less.

In order for the staff to authorize Electrosleeving beyond the two-cycle limitation, the remaining open issues from the May 20, 1998, NRC letter to Union Electric would have to be resolved. In addition, the growth of Electrosleeved cracks will need to be evaluated with an appropriate inspection technique at the end of the two cycles. Substantial additional work to characterize the performance of Electrosleeves under core damage accident conditions is still in progress. In the future, better ability to characterize flaws in Inconel tubing through Electrosleeves and better understanding of the performance of the Electrosleeves under severe accident conditions should be available. The staff will review the resolution of both of these issues, when submitted in the form of a TS change request, to determine whether Electrosleeves may remain in-service for longer than two cycles, and what restrictions are appropriate to their future application.