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SUBJECT: Forwards request for enforcement discretion from TS 3.3.1, Table 3.3-1, "RTS Instrumentation" & 3.3.2,Table 3.3-3, "ESFAS Instrumentation," permitting continued operation w/ SSPS inoperable for longer than allowed by TS 3.3.2.

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United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
SSPS/MLSB ACCIDENT ANALYSIS
REQUEST FOR ENFORCEMENT DISCRETION

Gentlemen:

Pursuant to 10 CFR Part 2, Appendix C, Carolina Power & Light Company (CP&L) requests that the NRC exercise enforcement discretion regarding compliance with Shearon Harris Nuclear Power Plant (SHNPP) Technical Specifications (TS) 3.3.1, Table 3.3-1, "Reactor Trip System Instrumentation," and 3.3.2, Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation." This request is made to permit continued operation with the Solid State Protection System (SSPS) inoperable for longer than allowed by TS 3.3.2. The enforcement discretion would also allow each reactor trip breaker to be bypassed for maintenance for longer than the 2 hours allowed by TS 3.3.1.

On February 2, 1995, the NRC notified CP&L of a potential vulnerability identified in the industry with respect to Solid State Protection System (SSPS) power supplies. Specifically, a scenario was identified for which the high energy line break (HELB) analysis for the Turbine Building had not analyzed the potential failure effect of anticipatory turbine trip channels on safety-related Class I SSPS logic power supplies.

CP&L has reviewed this scenario and determined that such a postulated failure could result in the failure of either train of the SSPS at SHNPP. Accordingly, a generic evaluation performed by Westinghouse and a probabilistic risk assessment performed by CP&L conclude that the Unit could safely operate until the commencement of the next scheduled plant shutdown. However, based upon our telephone conference call of February 7, 1995, we request that the NRC exercise enforcement discretion from the above TS requirements for a period of 10 days beginning Tuesday, February 7, 1995 at 1055. This will allow CP&L to implement and test a modification which will preclude the scenario described herein. The modification will separate the power source of the

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SSPS from the field input signals using a fuse and breaker coordination arrangement.

The justification for the request, including compensatory measures, a safety evaluation, and an evaluation of the potential impact on the public health and safety and the environment is enclosed.

Questions regarding this matter may be referred to Mr. R. W. Prunty at (919) 362-2030.

Sincerely,

LSR/lsr

c: Mr. S. D. Ebneter
Mr. S. A. Elrod
Mr. N. B. Le

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
SSPS/MSLB ACCIDENT ANALYSIS
REQUEST FOR ENFORCEMENT DISCRETION

REQUIREMENTS FOR WHICH DISCRETION IS REQUESTED

With respect to Technical Specification 3.3.1,Table 3.3-1,enforcement discretion is requested to permit the automatic trip logic to remain in the bypass condition for up to 12 hours per train in lieu of the specified 2 hours per train for the purpose of implementing and testing modifications for addressing the scenario described in this request.

With respect to Technical Specification 3.3.2,Table 3.3-3,enforcement discretion is requested to permit all Engineered Safety Feature (ESF) related items and Actions to be extended for the period requested (10 days), with individual actuation system logic allowed to remain in the bypass condition for up to 12 hours per train in lieu of the specified 2 hours per train, for the purpose of implementing and testing the modification.

CIRCUMSTANCES SURROUNDING THE SITUATION

A new scenario has recently been identified which could result in a failure of the Solid State Protection System (SSPS) to perform its intended safety function. The scenario involves a double-ended rupture steamline break in the turbine building. The break could occur in such a fashion that electrical conduits containing signals from the turbine stop valves and auto-stop oil pressure sensors that are input to the reactor protection system could be destroyed due to pipe whip and jet forces. This could cause these inputs to the SSPS to short out, causing a subsequent loss of SSPS power supply, resulting in the loss of one train of SSPS.

If a random single failure of the other train is assumed, no automatic protection functions would be available. A reactor trip would occur as the SSPS de-energizes, but Engineered Safety Features (ESF) actuations would not be available. Analog indications would remain operable and available to the control room operators. System actuations (e.g.,Safety Injection, Auxiliary Feedwater, Steamline Isolation) would not function; however, individual component actuators would remain operable. Thus, the operators would not be able to initiate Safety Injection or steamline isolation, but they could start individual pumps and close individual valves from the control room.

SAFETY BASIS FOR THE REQUEST

By letter dated February 3, 1995, Westinghouse provided a generic Justification for Continued Operation (JCO) to licensees in the Westinghouse Owners Group. CP&L has evaluated that JCO and determined its applicability to the Shearon Harris Nuclear Power Plant. The JCO Accident Evaluation is essentially restated below.

Accident Evaluation

The Westinghouse evaluation concentrates on the impact of the above postulated steamline break (SLB) scenario on core integrity. Mass and energy releases for both inside and outside containment are not impacted, since the break location is in the turbine building. To evaluate core integrity for this scenario, two initial conditions are considered: (1) the plant is at power with the control rods out of the core, and (2) the plant is at zero power with the control rods inserted, with the most reactive rod stuck in its fully withdrawn position.

From an at power condition, potential core damage is initially prevented by a reactor trip. When the input circuits short out as a result of the steamline break, one train of the SSPS de-energizes and a reactor trip occurs. A reactor trip at or very near nominal full power conditions is not a limiting condition with respect to the applicable acceptance criterion of DNB. Note that the Condition II acceptance criterion of DNB is conservatively applied even though the event is a Condition III/IV event with less restrictive requirements. Following the reactor trip at power, the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. The additional stored energy for a SLB initiated at power results in a less severe cooldown than a SLB at zero power. Therefore, the post-reactor trip time frame is bounded by the zero power analysis discussed below.

To address zero power concerns, two different four loop plants' zero-power, SLB core response analyses (similar to what is currently presented in a typical FSAR) were examined. The analyses assumed that all four steam generators blow down to the environment without operator action or automatic mitigation of any kind. The analyses retained standard FSAR conservatisms and assumptions as detailed below. The results demonstrate that not only is the DNB design basis met, but that the current FSAR licensing basis SLB core response analyses remain bounding.

The analyses assumed a double-ended rupture of the main steam header resulting in an effective break size of 5.6 square feet (1.4 square feet in each steam generator) corresponding to the total effective flow area of all four steamline flow restrictors. The analyses also assumed the following: hot zero power, end-of-life reactivity coefficients, no decay heat, all rods fully inserted with the exception of the most reactive rod fully

withdrawn from the core, steam generator level at the nominal program for hot zero power, no operator action, no automatic mitigation other than the passive cold leg accumulators, maximum auxiliary feedwater flow, and 100% nominal main feedwater flow. Note that assuming 100% of nominal feedwater flow for the entire duration of the transient conservatively bounds any short-term increase in feedwater flow as a result of the depressurization.

Typically, the RCS cooldown is weighted to one quadrant of the core and results in an asymmetric reactivity excursion. The fact that all four steam generators are contributing equally to the break results in a uniform reactivity transient in the core. Thus, even though the four steam generator blowdown transient results in a more severe RCS cooldown and depressurization, actuation of the cold leg accumulators and a more symmetric reactivity transient results in less-limiting peaking factors and DNBR value.

It should be noted since no automatic mitigation functions were assumed, the results of these analyses indicate that the same conclusion, i.e., that the current FSAR licensing basis SLB core response analysis would remain bounding, would be reached for other four loop plants. For three loop plants such as SHNPP, the event would be even less limiting since three loop plants have higher shutdown margins than four loop plants. Thus, the conclusions of the above discussed analyses would also apply to SHNPP.

Westinghouse also considered the effects of the postulated scenario on long-term core cooling in that the blowdown and subsequent dryout of all steam generators results in no available heat sink for the removal of decay heat and RCS residual heat. Westinghouse has concluded that for this aspect of the event sufficient heat removal is available through the initial blowdown and from residual feedwater flow via either the main feedwater pumps or the condensate booster pumps to preclude the need for the operators to take corrective action within the first 10 minutes of the event. Following 10 minutes, it is assumed that at least one motor driven pump would be available to remove any remaining stored energy and decay heat. Thus, the core will remain in a coolable geometry, pressures will be maintained below 100% of design pressures, and fuel cladding integrity will be maintained.

Analyses were also performed to evaluate the effects of an unisolated steamline break on the integrity of reactor vessels. These analyses were based primarily on the results and information from WCAP-10319 (1983). The analyses took into consideration the resulting RCS temperature and pressures given an unisolatable steamline break with a total loss of the SSPS. The results of the Pressurized Thermal Shock (PTS) evaluation concluded that the postulated steamline break is an insignificant contributor to the risk of significant flaw extension in the reactor vessel.

Although not directly credited in the above evaluation, it is assumed that the operators would be following the applicable Emergency Operating Procedures to isolate the steam

generators, if possible, and to insure long term heat removal capability, etc. This evaluation would also be applicable to three loop plants such as SHNPP. Evaluation of the postulated SLB scenario concludes that non-LOCA safety analyses for all Westinghouse plants do not violate the applicable acceptance criteria, and remain bounded by the FSAR licensing basis.

Probabilistic Evaluation

The risk of core damage from the steamline break-SSPS failure scenario is estimated at 3.3×10^{-8} per reactor year, more than three orders of magnitude less than the annual core damage frequency for Harris due to all causes. The event frequency is only slightly above the 10^{-8} truncation limit used in the Probabilistic Safety Analysis (PSA).

This analysis evaluates the probability of a severe (core damage) accident due to a postulated scenario involving a failure of steamline piping in the turbine building. It is postulated that a large steamline piping failure occurs in the vicinity of the SSPS electrical components in the turbine building, such that the resultant pipe whip and jet impingement damage these components, resulting in an electrical fault which disables one train of the SSPS. This failure causes a reactor trip due to failure of the SSPS train. A single failure assumed in the remaining SSPS train results in a loss of actuation capability for steamline isolation, safety injection actuation and auxiliary feedwater (AFW) actuation. It is further assumed for this analysis that main feedwater fails, and no automatic AFW actuation occurs after the main feedwater pumps trip. The resulting event is a large secondary line break with failure of steamline isolation, no safety injection and no secondary heat removal after steam generator dryout.

The sequence of events leading to core damage for this scenario is:

- large steamline break in vicinity of SSPS components
- single failure of one train of SSPS
- assumed failure of main feedwater to continue operating leading to loss of secondary side heat removal
- failure of operator to manually actuate AFW

The Harris PSA assumes a frequency for secondary line breaks of 2×10^{-2} per reactor-year. This event includes both feedwater and steamline breaks, and is dominated by events involving small breaks, especially due to stuck open relief valves, pump seal failures, etc. The frequency of a large steamline break is not significant compared to these smaller breaks. The frequency of a large piping failure in the RCS (breaks above 5") is 3.3×10^{-5} per reactor-year for the Harris Plant, based on an analysis using EPRI methodologies for calculating RCS piping failures. Since the steamline piping is not of the same quality class as the RCS, a two order of magnitude increase is applied to the

frequency, to arrive at a 3.3×10^{-3} per reactor-year frequency for a large steamline break. This number is judged to be very conservative. The Westinghouse analysis uses a 0.1 value for the break to occur at a location which could damage the SSPS components. This is also judged to be very conservative. The final frequency of a large steamline break in a critical location for SSPS failure is then 3.3×10^{-4} per reactor-year.

The probability of failure of one train of SSPS is estimated by Westinghouse to be 0.01. A review of the Harris PSA shows that each SSPS train has a failure rate of about 0.01, and so a probability of SSPS failure of 0.01 will be used.

From the Westinghouse analysis, at least ten minutes are available for operator action. The operator would follow EOPs after the reactor trip, and would have indications of low steamline pressure and low RCS pressure, requiring actuation of SI. Although manual actuation of the SSPS would be disabled, manual actuation of individual components, as directed by immediate actions in the EOPs, would be operational. The operator would also have indications of loss of secondary cooling (low steam generator levels and eventual RCS heatup), requiring actuation of feed-and-bleed cooling. The indications requiring actuation of equipment are judged to be at least as compelling as was assumed in developing Human Reliability Analysis (HRA) values for manual start of equipment (following automatic signal failure) for the PSA. The HRA value for failing to actuate AFW is 5.3×10^{-3}. A conservative 10^{-2} value will be used to further bound this analysis.

Using these values, the total event frequency (F) is estimated as:

F = (steamline break at SSPS components)(SSPS train failure)(operator failure to start AFW)

$F = (3.3 \times 10^{-4})(0.01)(10^{-2}) = 3.3 \times 10^{-8}$ per reactor-year.

This is more than three orders of magnitude less than the nominal core damage frequency from all causes of about 5.9×10^{-5} per reactor-year, even with the conservative assumptions for initiating event frequency and human reliability. The event frequency is only slightly above the 10^{-8} truncation limit used in the PSA.

COMPENSATORY MEASURES

The following compensatory measures will be taken to provide additional assurance that the public health and safety will not be adversely affected by this enforcement discretion request:

1. The design change will be performed on only one train of the SSPS at a time. This will provide assurance that at least one train of SSPS would perform its required function to mitigate the consequences of postulated accidents.

2. Optional train-related maintenance and surveillance testing will be minimized, and suspended when possible, until implementation of this modification.

3. Plant evolutions significant to risk will be avoided.

4. Operations orders have been prepared describing the condition and the proper implementation of the emergency procedure for responding to a MSLB that affects the SSPS. Operators on shift are being briefed on the scenario of concern, how to identify the scenario, and mitigating actions to take.

5. Activities on the turbine deck that could result in damage to the steam lines (such a movement of loads over the high pressure turbines) will be restricted until implementation of the above mentioned design change.

6. Lessons learned from the Salem modification installation will be reviewed and incorporated into the SHNPP modification implementation.

JUSTIFICATION FOR NONCOMPLIANCE DURATION

Based upon our review of the generic Justification for Continued Operation provided by Westinghouse, as well as the results of the plant specific probabilistic risk assessment of the scenario, CP&L concludes that the Unit could safety operate until the commencement of the next scheduled plant shutdown. However, we request that the NRC exercise enforcement discretion for 10 days beginning Tuesday, February 7, 1995 at 1055. This will allow CP&L to implement and test a modification which will preclude the scenario described herein.

SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination. The bases for this determination are as follows:

1. The proposed enforcement discretion will not involve a significant increase in the probability or consequences of an accident or malfunction previously evaluated. The probability of a MSLB accident is not affected by the proposed enforcement discretion. The only equipment failure potentially affected is the failure of one train of SSPS. Failure of the SSPS as the initiating event or failure of an SSPS train following any accident other than a steamline break is unaffected. A plant specific risk assessment was performed which indicates that the increase in SSPS failure probability resulting from the proposed enforcement discretion is insignificant. The consequences of a malfunction of the SSPS due to a MSLB in the turbine building coincident with a single active failure of one train of the

SSPS were evaluated by Westinghouse. The evaluation concluded that the DNBR limits would be satisfied. Long term cooling can be assured by operator action to restore one motor-driven AFW pump within 10 minutes. NUREG-0800 allows operator action to be credited in mitigating the consequences of an accident. CP&L has reviewed the operator response to a MSLB scenario and concludes that the operators would be capable of mitigating the consequences of this MSLB in adequate time to prevent core damage. Pressurized thermal shock has also been shown to be insignificantly affected by this scenario.

2. The proposed enforcement discretion does not create the possibility of a new or different kind of accident from any accident previously evaluated. A MSLB has been evaluated in the FSAR. The proposed enforcement discretion does not create the possibility of additional accident initiating events.

3. The proposed enforcement discretion does not involve a significant reduction in the margin of safety. A risk assessment was performed that determined that the probability of a MSLB that disables one train of SSPS coincident with a single active failure of the other SSPS train during the period of the enforcement discretion was insignificant. CP&L's review of the operator response to a MSLB scenario concludes that the operators would be capable of mitigating the consequences of the MSLB in adequate time to prevent core damage. Further, based upon the Westinghouse evaluation of a MSLB without any SSPS or operator action available, the DNBR limits would be satisfied.

ENVIRONMENTAL EVALUATION

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. Carolina Power & Light Company has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

1. As demonstrated in the above significant hazards analysis, the proposed enforcement discretion does not involve a significant hazards consideration.

2. The proposed request for enforcement discretion does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. DNBR and PTS limits are not exceeded and long term core cooling is assured. The proposed change does not introduce any new equipment nor does it require any existing equipment or systems to perform a different type of function than they are presently designed to perform.

3. The proposed request for enforcement discretion does not introduce any new equipment, change any operating practices, or result in any new surveillances or testing. It therefore, does not result in a significant increase in individual or cumulative occupational radiation exposure.

PLANT NUCLEAR SAFETY COMMITTEE REVIEW

This request for enforcement discretion and its basis have been reviewed by the SHNPP Plant Nuclear Safety Committee, and that Committee concurs with this written request.

CONCLUSION

Based upon the Westinghouse evaluations that have shown that the accident analysis criteria continues to be met and the plant-specific probabilistic safety analysis which shows that the probability of the occurrence of this scenario is sufficiently low, the postulated scenario is not expected to present an undue risk to the public health and safety. Carolina Power & Light Company is fully aware of this potential scenario and is taking appropriate contingency actions as described above. Based upon this and the Safety Evaluation provided above, we conclude that the Shearon Harris Nuclear Power Plant could continue to operate until the next scheduled plant shutdown. However, to address this issue expeditiously, CP&L intends to implement a modification by February 17, 1995.