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September 22, 1980

Mr. Robert Bernero, Director
Probabilistic Analysis Staff
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Oconee Unit 3
Draft of RSSMAP Report

Dear Mr. Bernero:

Your letter of September 3, 1980 enclosed the draft report on the Oconee RSSMAP study, performed by the Probabilistic Analysis Staff and its contractors, for our review and comment. We have been able to conduct only a limited review of this draft because of the short time made available to us. Nevertheless, we have found a number of areas in the study and in the report itself where changes would be desirable and perhaps necessary. These areas of concern include the validity of the system success criteria, correctness and appropriateness of the assumptions made in the systems analysis, and the applicability of the comparison of the results of the Oconee study to those of the RSS reference plant. Our comments are summarized below and elaborated in the Attachment to this letter.

In the area of the system's success criteria, we found that the criteria used for the emergency core cooling were incorrect and that the success criterion used for the emergency feedwater system needs to be modified.

In the systems analysis area, there exist discrepancies between the assumed and actual configurations of the HPI and LPI systems. Also the assumed values of failure probabilities for manual actions in a number of instances we believe are overly conservative (HPI cooling, steam generator cooling by HHASWS, reestablishing AC power, etc.).

It is apparent that in some instances an attempt was made to utilize Oconee specific component reliability (reliability of the turbine-driven

emergency feedwater pump and the reliability of the Keowee units). Our review of the applicable failure data indicates that the failure and demand data were not properly interpreted, and as such the unavailabilities for the affected systems were significantly overestimated.

We understand that in the RSSMAP study the RSS methodology was modified in several critical areas (common mode failure contribution, human reliability, etc.). Thus the indicated difference in the risk and severe core melt probabilities between the Oconee and the RSS reference plant does not provide a valid comparison.

The problem areas and discrepancies summarized above and elaborated in the Attachment would have a significant impact on the calculated frequencies of each of the dominant sequences, except perhaps the Event-V sequence, and as such warrant appropriate reanalysis. We suggest that, inasmuch as possible, the draft report be revised taking into account the review comments provided herein.

We recognize that the Oconee RSSMAP study had to rely primarily on FSAR-type information, which is outdated in several areas, and did not have the benefit of a more complete information base on the actual plant characteristics in several important aspects. A more thorough characterization of the Oconee accident sequences and plant risk is expected by way of the Duke/NSAC Oconee PRA program. The Oconee RSSMAP study would certainly serve as a valuable reference and road map for the Oconee PRA program.

It should be pointed out that the Oconee RSSMAP study and our earlier auxiliary feedwater system reliability study have uncovered two areas of the Oconee emergency feedwater system where further improvements are possible and desirable. Design modifications have been initiated to eliminate the AC dependencies in the turbine-driven EFWS train. As a result of these modifications, flow of cooling water to the turbine oil cooler and to the pump cooling jacket will be by gravity flow from the high pressure service water system. This modification should significantly reduce the unavailability of the EFWS for transients involving loss of offsite power or loss of all AC power. With regard to the Event-V sequence, Duke is evaluating measures to reduce the risk of this event and will implement appropriate procedure changes and/or modifications to reduce the risk to an acceptable level.

In summary, we consider the report to be flawed to such a degree that correction of certain portions of the report should be considered prior to publication. To resolve these comments, we suggest that a meeting be held at a mutually agreeable time.

Any questions or comments regarding the matters discussed in this letter and attachment may be directed to P. M. Abraham of our Project Coordination and Licensing Section (phone: (704)373-4520).

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

PMA:vr
Attachment

DUKE POWER COMPANY COMMENTS

ON

DRAFT OCONEE 3 RSSMAP REPORT

September 22, 1980

DUKE POWER COMPANY COMMENTS ON OCONEE RSSMAP REPORT

The following comments primarily address the assumptions and analyses in the Appendices, the source of basic information, but are also applicable to pertinent sections of the main body of the report.

I. Appendix A1. LOCA Event Trees.

2.1.4 The success criteria for emergency core cooling identified on page 14-57 of the FSAR and used in RSSMAP have been revised as the result of a more recent analysis. The currently applicable criteria, based on Appendix K requirements and documented in BAW-10103, Rev. 3A, are listed below:

LOCA	Equivalent Diameter	Success Criteria
Large Break (A)	D > 10"	2/2 CFT and 1/3 LPI Pumps
Small Break (S_1)	4" < D $\leq$ 10"	2/2 CFT and 1/3 HPI Pumps and 1/3 LPI Pumps
Very Small Break (S_2)	D $\leq$ 4"	1/3 HPI Pumps

Therefore, it is not necessary to consider a specific break spectrum in the range of 10" < D $\leq$ 13" In addition, the success criteria used are overly conservative and the emergency core cooling unavailability will be reduced, thereby reducing the estimated frequencies of sequences involving the term D.

2.2.1 Per the preceeding discussion, only three break sizes need to be considered, eliminating the 10" < D $\leq$ 13" category.

2.2.2 The RPS is required to operate for S_1 and S_2 LOCA's, but not for A LOCA's as redefined above.

2.2.8 The success criteria for ECR are listed below:

LOCA	Functional Success
A	1 of 3 LPRS
S_1	1 of 3 LPRS
S_2	1 of 3 LPRS and 1 of 3 HPRS

3.0 Per the preceeding discussion, a stuck-open RCS relief valve results in an S_2 LOCA rather than S_3 LOCA.

Table A-1 should be revised to reflect the revised success criteria discussed above.

II. Appendix A2. Transient Event Tree

2.1 Per the redefinition of LOCA categories, RCS integrity is required to prevent a small small (S_2) LOCA.

2.1.2 The RCS heat removal requirements as stated are unduly conservative, and the following evaluation demonstrates that flow from any one of the three EFW pumps to either steam generator is sufficient.

Using the August, 1979 ANS Decay Heat Standard and assuming an EFW enthalpy of 61 Btu/lb (corresponding to 90F and 1000 psi), the EFW flow requirements following shutdown are listed below:

Time (Sec)	Power Function	Flowrate (gpm)
60	0.0342	531
80	0.0322	500
100	0.0308	479
150	0.0283	440
200	0.0267	415
400	0.0232	361
600	0.0213	331

The demand for EFW flow occurs when the initial steam generator inventory boils off to the minimum level. At this time the capacity of one motor-driven pump (500 gpm) is adequate for RCS heat removal. This will reduce the unavailability of the EFW system, and consequently the frequencies of sequences involving the term L.

Two additional means of removing RCS heat were not discussed explicitly. If off-site power is available, one hotwell pump and one condensate booster pump provide sufficient flow if steam generator pressure is reduced using the turbine bypass valves. The original auxiliary service water pump is also still available, requiring opening of the manual atmospheric dump valves to depressurize the steam generators to approximately 60 psig.

As reported in BAW-1610, the maximum RCS pressure expected for an event involving failure of the RPS is 3600 psig, rather that 4000 psig.

2.2.3 The unavailability estimate of the PCS during T_2 transients is unrealistically high for Oconee. To obtain the necessary main feedwater flow prior to SG dryout, only one train of the hot-well - condensate booster - main feedwater pumps combination is necessary. In the event the PCS recovery is unsuccessful prior to SG dryout, the hotwell-booster pump combination can be utilized since the resulting SG pressure is low, and further the turbine bypass valves can be controlled to maintain SG pressure within the HWP-CBP flow capability. The Oconee turbine driven main feedwater pumps can be supplied with motive steam via the station auxiliary steam header.

A survey of recent Oconee operating experience for 1979 and 1980 identified nine reactor trips involving feedwater transients. In all nine occurences, the main feedwater system was either available throughout the transient or was recovered within 30 minutes of the reactor trip. This operating experience supports the contention that a PCS non-recoverability probability of 1.0 grossly exagerates the unavailability of the PCS during transients other than T_1 transients.

Based on the above considerations, it is inappropriate to assume a value of 1 for the PCS unavailability during T_2 events. Rather, a value of 10^{-2}, consistent with the operating experience, would be appropriate. This change would reduce the presently calculated probability of sequences involving the term M.

2.2.4 The discussion should identify EFW capabilities as two 100% motor-driven pumps and one 200% turbine driven pump.

2.2.7 The frequency of the PZR relief valves to reseat after being challenged is assigned the same value (10^{-2}) for the PORV and the safety values. We believe that the value for the safety valves is much smaller (of the order of 10^{-4}). With respect to the PORV, operator action to close the PORV block value should be considered in light of the recent changes in operator training and procedures and the implemenation of direct PORV position indications. Considering the smaller frequency of safety value failure and the changes in PORV isolation capability, it is suggested that a much smaller value for Q, perhaps in the range of 10^{-4} - 10^{-3} would be appropriate. This change would reduce the currently calculated frequencies of sequences involving Q by a factor of 10 to 100.

The unavailability of feed and bleed cooling during transient sequences involving failure of the PCS, EFS, and HASWS is overestimated. HPI cooling in the absence of SG cooling is explicity required by the emergency procedures. Because of the RCS saturation alarm and the obvious indications of inadequate SG cooling, and considering that a time interval of greater than 20 minutes is available to initiate this

function, we believe that the unavailability of this function is of the same order as that of the ECC recirculation function. Furthermore, continued boil-off of the RCS inventory through the PZR relief values would eventually actuate the ESF on high RB pressure.

3.0 The stuck open relief safety valves result in S_2 LOCA's rather than S_3 LOCA's.

III. Appendix B1. Emergency Power System

2.1.1 At least one of the two Lee combustion turbines is started and energizes the dedicated 100 KV line to Oconee whenever one of the Keowee units is unavailable. In the event the Keowee outage is an extended one, a load from Oconee (normally a 4160 volt bus) is placed on the Lee turbine for operating reliability.

5.2 The unavailability calculated for the Keowee units based on the number of tests and failures appears to be too high. It was apparently assumed that only a monthly test of each Keowee unit was made, yielding 168 tests. However, seven annual tests have been performed in addition to system demands upon the Keowee units to supply power to the grid. Properly accounting for Keowee startup demands from all sources indicates that greater than 2500 demands have occurred. Twelve instances have been identified wherein a unit failed to deliver power upon demand, regardless of the source of the demand. Of the twelve instances, seven involved a unique failure of a particular component over a five month period. The problem is belived to have been rectified. This type of failure has not reoccurred in the succeeding two years and may justifiably be counted as a single failure. Thus, the Keowee failure per demand ratio can be conservatively calculated as 6/2000.

On a qualitative basis, the Oconee EPS reliability is equal to or better than that of many other nuclear stations. Alternate power sources are available to an individual Oconee unit from the other nuclear units at the station, from the system grid, from the near site Keowee units, and from the combustion turbines at the Lee Steam Station. The hydroelectric generators at the Keowee facility have inherently simpler design and operating characteristics relative to diesel generators and thus represent a more reliable backup power source. Furthermore, the Keowee units are frequently called upon to supply power to the system grid and therefore problems are more likely to be detected and corrected prior to emergency use. Based upon these considerations, the AC power non-recovery probability (within ½ to 3 hours) of 0.1 - 0.5 used in the report is unrealistic.

IV. Appendix B3. Reactor Protection System

5.2 $Q(RPS) = 2.6 \times 10^{-5}/RY$, not $2.6 \times 10^{5}/RY$

Table B3-1 Two reactor trip setpoints are incorrectly listed. The correct setpoints are identified below:

Over Power	105.5% of rated power
RC Pressure	2300 psig - High

V. Appendix B6. Low Pressure Injection System

2.1 During normal operation, motor-operated valves LP-21 and LP-22 in the LPIS suction lines from the BWST are left open. This will reduce somewhat the unavailability of the LPIS.

Although it is not important to the results, the notation for the LPIS pumps and coolers refer to Oconee 1 components. The correct notation for Oconee 3 is LP-P3A, LP-C3A, etc.

5.1 The correct success criterion for LPIS for both A and S_1 LOCA's is one out of two pump trains.

5.2.1 Since the LPIS success criterion is the same for LOCA's A and S_1, only one Boolean equation is required, i.e., Equation B6-1.

5.2.2 The LPIS unavailability for failure of 2 of 2 trains is recalculated for the case when MOV's LP-21 and LP-22 are normally open.

for LP-21 plugging	1×10^{-4}	
LP-22 operator error	3×10^{-4}	(Consistent with LP-28)
Q_2 total	4×10^{-4}	

$\therefore$ B: LP-22 + LP-30 $\quad 4 \times 10^{-4} + 10^{-4} = 5 \times 10^{-4}$

C: LP-21 + LP-29 $\quad \simeq 5 \times 10^{-4}$

The unavailability of the LPIS is therefore reduced from 2.6×10^{-3} to 2.1×10^{-3}, so the change is not very significant.

Table B6-4 should be deleted.

VI. Appendix B7. Low Pressure Recirculation System

5.1 Successful ECR requires one LPRS train for A and S_1 LOCA's.

5.2.1 The Boolean equation for LPRS failure excludes some terms included in LPIS since success of LPRS is important only given success of LPIS. Therefore, the terms from the LPIS equation which are included here do not have the same unavailabilites, and should be lower.

Also, although the discussion in 2.1 includes the third LPIS pump, no credit is taken for its availability in the analysis.

VII. Appendix B8. High Pressure Injection System

2.1 The electric power for each of the three HPIS pumps is supplied from a different 4160 volt bus. It appears that this discussion should refer to the digital ES channels which actuate the three pumps.

2.2 The injection valve in the B train (HP-27) is normally left open with power to its motor operator removed. This should reduce the unavailability of this train somewhat. A cross connection is available between all three injection lines, with isolation by normally closed motor-operated valves HP-409 and HP-410. The injection points are downstream from the normal isolation valves, HP-26 and HP-27, allowing operator action to assure two trains of HPIS flow in the event of failure of the injection valves or an HPI pump. This also should reduce the unavailability of the HPIS.

5.2.1 The HPIS unavailability due to pump testing is incorrectly calculated. At Oconee, the A and B HPI pumps are used alternately to supply normal makeup seal injection. Test procedures for the C pump specify that it be used to supply normal makeup by closing HP-27 (refer to Figure B8-1) and opening the cross-tie valves from the B injection line, HP-116 and 117 (not labeled in the figure). Thus, this pump is available should an HPI actuation signal occur during testing of that pump. Therefore, HPI pump testing does not contribute to the HPIS unavailability.

VIII. Appendix B9. High Pressure Recirculation System

2.1 The discussion concerning electric power contains the same error identified above, i.e., three separate buses supply power to the three respective pumps.

IX. Appendix B10. Engineered Safeguards Protection System

2.2 In response to NUREG-0578, reactor building isolation is actuated by channels 1 and 2 to provide isolation on either low reactor coolant pressure or high containment pressure.

X. Appendix B11. Containment Spray Injection System

2.2 Valves LP-21 and LP-22 are normally open and are therefore not required to change positions on ES 7 and 8 actuation. This should reduce the unavailability of the CSIS.

5.0 The dominant failure contributor of the CSIS during transients and small break LOCA events was treated to be the failure of the operator to start the system whenever the CSIS is manually bypassed. Resetting the applicable ESFAS channels would still maintain the system in the safety mode. Deliberate operator action is required to bypass an automatic safety function and is not permitted unless it is confirmed that the plant mode does not require that function. Even under bypass conditions,

the high reactor building pressure alarm would alert the operator to activate the sprays. Therefore, the assignment of a higher probability for the CSIS failure during transients and small break LOCA events does not seem to be necessary.

XI. Appendix B13. Emergency Feedwater System and High Head Auxiliary Service Water System

1. Since 1973, 158 tests of the turbine driven emergency feedwater pumps have been performed. A review of the test results identified four incidents wherein the pump failed to start. This data supports a pump unavailability probability of 2.5×10^{-2} which is more than three times lower than the value used in the report.

2. The auxiliary service water system cross-ties from the other units are still available, in addition to the new high head system. Thus, two additional backup systems are available.

3. In addition, as discussed in the comments for Appendix A2, the correct success criterion for EFW is one of the three pumps, or a flowrate of 500 gpm.

4. The unavailability of the HHASWS is est: ated at 0.1 based on human error. This seems unrealistically high considering the fact that it is a dedicated system and further tha: the operators will be dedicated to the SSF. It is expected to yield fairly high availability, on the order of that for the ''RS.

XII. Appendix B15. Reactor Building Cooling Systems

5.2.2 The failure of the RBCS was estimated for cases with and without AC power initially available, although these cases were not investigated for other systems.

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November 3, 1981

Mr. Thomas M. Novak
Assistant Director for Operating Reactors
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Oconee Unit 3
RSSMAP Report

Dear Mr. Novak:

The following information is submitted in response to your letter of September 8, 1981 concerning the Reactor Safety Study Methodology Application Program (RSSMAP) study of Oconee Unit 3. Duke Power Company previously (by our letter of September 22, 1980) provided comments on an earlier draft report of this study. We note with appreciation that some of our comments were considered in the preparation of the final version of the report. Specific comments on the conclusions of the study reported in Section 6.3.1 and some discussion on changes in plant systems and procedures implemented at Oconee subsequent to the RSSMAP study with potential positive impact on the RSSMAP estimated probabilities and consequences of accidents are provided in the following paragraphs.

With regard to the frequency of an interfacing system LOCA event, we have instituted a program for periodic leakage testing of the check valves of interest and further have ceased the stroke testing of the normally closed MOV's LP-17 & -18 at operating conditions. (Stroke testing of these values is done only at cold conditions.) Since the leak-leak failures are essentially eliminated by the periodic leak testing program and by elimination of the MOV stroke testing at operating conditions and since the leak-rupture and rupture-rupture failures are significantly small with the normally closed configuration of the MOV's, the event V is now believed to be a non-significant risk contributor.

Plant modification has been completed to eliminate the AC dependency of the turbine driven emergency feedwater pump. With this modification, the availability of the emergency feedwater system during accidents involving loss of offsite power or loss of all AC power has improved. Consequently, the frequency of core melt accidents initiated by or involving these events would be less than that estimated in the RSSMAP study.

Two changes which are outside the scope of the RSSMAP study and which came to our attention in conjunction with the ongoing NSAC-Duke Oconee PRA program are being implemented now. One is a change in the emergency procedures to deal with a situation in which the LPI pumps could be running at shutoff head for an extended period of time. Such a situation is postulated to occur during

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certain small break LOCA and severe steamline break events if the RCS does not depressurize sufficiently below the LPI actuation setpoint. Although the operators are aware of the need to secure the LPI pumps within reasonable time under this situation, the existing emergency procedures do not include this requirement. A change in the applicable procedures is now being implemented to include the necessary guidance. The other change pertains to two ICS simulator relays. A postulated spurious energization of these relays could lead to a feedwater transient (resulting in a reactor trip) and the turbine bypass valves failing closed. A modification of the system to deactivate this circuitry is being implemented. The interim results of the Oconee PRA are being monitored to assure early identification of any important risk outliers. At this point, no additional items have been identified which merit consideration in the near term.

A number of the post-TMI changes to plant systems and procedures have contributed to improved safety both with respect to probabilities and consequences of accidents. Among the measures contributing to reduced probabilities of accidents are:

(a) Modification to the turbine EHC System to reduce the frequency of turbine-reactor trips.

(b) Modifications to the main feedwater system to minimize the occurrence of feedwater transients.

(c) Changes in the control system power supply to minimize the occurrence of power supply failure induced transients and to better cope with such events.

(d) Modification of the emergency feedwater system initiation, control, and indication functions for better reliability and performance.

The following other post-TMI efforts, which are in various stages of implementation, are considered to effect further reduction in the probabilities and consequences of accidents.

(a) Renewed vigilance and searching reviews now being conducted on operational occurrences through the operating experience evaluation program.

(b) Control room design review and incorporation of safety parameter display system.

(c) Improved operator training, development and implementation of improved procedures and the utilization of shift technical advisors.

(d) Implementation of RCS high point vents, post-accident sampling panel, and dedicated hydrogen penetrations.

(e) Implementation of PORV/PSV position indicator and RCS subcooling monitor.

(f) Implementation of accident monitors and expanded emergency planning programs and facilities.

We are not certain whether the conclusion reached in the Oconee RSSMAP study regarding hydrogen burning and the associated impact on containment is valid. There is reason to believe that the Oconee containment failure pressure is much higher than that assumed in the RSSMAP study (183 psia versus 133 psia). Furthermore, the MARCH code treatment of hydrogen in regard to its generation in the core, accumulation in the containment, and degree of burn in the containment is generally recognized to be very conservative, particularly for small break and transient induced core melt events. A more realistic analysis of the containment accident process is expected in the NSAC-Duke Oconee PRA analysis.

It is our impression that the Oconee RSSMAP study has been a very worthwhile undertaking. Although there are some limitations in this study, it still provides some useful insights into the dominant accident sequences and their contributing factors.

Very truly yours,

William O. Parker, Jr.

PMA/php

cc: Mr. Robert F. Bernero, Director
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