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U. S. Nuclear Regulatory Commission
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Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Responses to Request for Additional Information Concerning
Proposed License Amendment Request to Extend the Completion Time
for an Inoperable Low Pressure Injection Train from 72 Hours to 7 Days,
License Amendment Request No. 2001-012

On February 11, 2003, a conference call was held between Mr. Lenny Olshan and Mr. Millard Wohl of the Nuclear Regulatory Commission (NRC) and members of Duke Energy Corporation's (Duke) technical staff, to discuss additional NRC questions in regards to the above mentioned License Amendment Request. At the completion of the meeting, Mr. Olshan requested that Duke formally submit these responses to the Staff. The enclosure to this letter provides Duke's written responses.

If there any questions regarding this submittal, please contact Stephen Newman of the ONS Regulatory Compliance Group at 864-885-4388.

Very truly yours,

R. A. Jones, Vice President
Oconee Nuclear Site

Enclosure: Responses to Request for Additional Information

A001

cc: Mr. L. N. Olshan, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
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61 Forsyth St., SW, Suite 23T85
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Mr. M. C. Shannon
Senior Resident Inspector
Oconee Nuclear Station

Mr. Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

R. A. Jones, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Facility Operating License Nos. DPR-38, DPR-47, and DPR-55, for Oconee Units 1, 2, and 3 respectively; and that all the statements and matters set forth herein are true and correct to the best of his knowledge.

R. A. Jones, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 19 day of February, 2003

Connie M. Breazeale
Notary Public

My Commission Expires:

12/19/12

SEAL

ENCLOSURE

DUKE RESPONSES TO RAI CONCERNING
PROPOSED LICENSE AMENDMENT REQUEST TO EXTEND THE COMPLETION TIME FOR AN INOPERABLE LOW PRESSURE INJECTION (LPI) TRAIN FROM 72 HOURS TO 7 DAYS
OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

Question 1:

Provide a discussion of the risk impact of external events per RG 1.177.

Response:

The risk impact of external events was determined quantitatively in the original analysis and was discussed briefly in the October 24, 2002 submittal to the NRC where it said, "......the incremental or increase in CDF did not change appreciably when external events were included." The same statement was made for LERF in that submittal. The external events solution contained a seismic, tornado, fire, and external flood contributions. The risk numbers are as follows:

CDF & LERF values are per yr.	Internal Events Only	Internal and External Events (includes seismic)
Base Case CDF	2.61E-05	8.92E-05
CDF for 7 day CT (mean mnt. duration)	2.63E-05	8.94E-05
CDF for 7 day CT (full mnt. duration)	2.64E-05	8.96E-05
CDF Increase (mean mnt. duration)	2.0E-07	2.0E-07
CDF Increase (full mnt. duration)	3.0E-07	4.0E-07
Base Case LERF	1.07E-07	5.864E-07
LERF for 7 day CT (mean mnt. duration)	1.027E-07	5.866E-07
LERF for 7 day CT (full mnt. duration)	1.074E-07	5.869E-07
LERF Increase (mean mnt. duration)	2.0E-10	2.0E-10
LERF Increase (full mnt. duration)	4.0E-10	5.0E-10
Conditional CDF, 1 LPI train OOS	4.37E-05	1.07E-04
Conditional LERF, 1 LPI train OOS	1.30E-07	6.091E-07
ICCDP for 7 day duration (unitless)	3.4E-07	3.4E-07
ICLERP for 7 day duration (unitless)	4.4E-10	4.4E-10

The following external events are included in the model:

Seismic

The Oconee PRA plant fault tree contains all evaluated internal and external events except for the seismic initiator. The seismic fault tree is generated and solved separately from the main PRA fault tree because the seismic core damage frequency (CDF) is calculated using Monte Carlo techniques. The core damage sequences that dominate the seismic results are the station blackout sequences. From a plant response perspective, these sequences look very much like the sequences that result from tornadoes and LOOP initiated transients since the same systems are required to prevent core damage. These core damage sequences are characterized by the loss of all engineered safeguards systems as a result of the station blackout. As a result, the failure of individual components or individual train failures in mechanical systems- such as the LPI system- has little or no influence on the seismic results. The exclusion of the seismic events greatly simplifies the analysis without a significant loss of accuracy in the calculated CDF. The Oconee seismic contribution is 3.92E-05/yr.

Tornado

The dominant tornado core damage sequences involve a loss of off-site power, a failure of the Standby Shutdown Facility (SSF), and a loss of both Emergency Feedwater (EFW) and Station Auxiliary Service Water (ASW) due to tornado damage to equipment in the Oconee Turbine or Auxiliary Buildings, or damage to the Keowee Emergency Power System. The leading causes of SSF failure are failure of the SSF Diesel Generator to Run, tornado damage to SSF related piping and cabling located in the West Penetration or Cask Decon Rooms, and failure of operators to align the SSF systems in time.

Unavailability of an LPI train has a negligible impact on tornado risk because of the vulnerability of the 4kV Auxiliary Power System in the Turbine Building. The 4kV power system provides AC power to the LPI pumps as well as to other pumps that are needed for cooling water.

External Flood

The dominant sequences for external flood involve a failure of on-site power sources and a failure of the SSF due to flooding, random SSF failures, or failure of operators to align the SSF systems. Unavailability of an LPI train has a negligible impact on external Turbine Building flood risk for the same reason as stated above.

External Fire

Similar to external flood, the dominant sequences involve a failure of on-site power sources and a failure of the SSF due to random SSF failures, or failure of operators to align the SSF systems due to fire damage to equipment in the Turbine Building. Again, unavailability of an LPI train has a negligible impact on external fire risk for the same reason as stated above.

Question 2:

Discuss the four (4) peer review findings received by Oconee during the B&W Owners Group (BWOG) peer review that were classified as "important and necessary" to address the technical adequacy of the PRA, the quality of the PRA, and the quality of the PRA update process.

Response:

- Item 1 - Revision 2 of the PRA did not include a method for methodically evaluating the dependence among human actions. The human reliability analysis (HRA) for Revision 3 had not progressed to the extent that review of the evaluation of human reliability dependencies was possible.

  The methodology to be implemented for Revision 3 was found acceptable based on a review of its implementation at another Duke nuclear station. For Oconee, a review of the cut sets that included LPI events did not identify HRA combinations. Therefore, this is not a significant issue for the LPI change requested.

- Item 2 - The interfacing systems loss of coolant accident (ISLOCA) frequency in Rev. 2 reflects a point estimate of cut sets of valve failure modes that have very large uncertainties. A point estimate does not represent a reliable estimate of the mean ISLOCA frequency due to the propagation of uncertainties through the ISLOCA cut sets.

  PRA Rev. 3 ISLOCA analysis approach was revised to determine the mean ISLOCA frequency in addition to the point estimate. This issue has no impact on the LPI change requested.

- Item 3 - Key contributions to LERF in Rev. 2 may have been underestimated. Primary issues identified relate to the quantification of the Steam Generator Tube Rupture (SGTR) event tree and with the mapping of SGTR cut sets to an appropriate plant damage state.

  The Oconee estimated LERF, which is dominated by the (ISLOCA), would not be significantly impacted by the proposed change. ISLOCA goes to core damage which the LPI system cannot mitigate. Therefore there is no impact on the LPI change requested.

  The dominant sequences requiring LPI in the SGTR analysis are those where cooling down for normal decay heat removal is the desired end state. Failure to establish normal decay heat removal is dominated by failures of the valves that must open to align the Reactor Coolant System (RCS) to the LPI pump suction and not the LPI pump trains.

Early Containment Failure is a relatively small issue for PWRs with large dry containments. The extended LPI CT has no impact on the conditional early containment failure probability. This contribution to LERF would be expected to be roughly proportional to the change in CDF and a small fraction of its value.

- Item 4 - The completeness in modeling common cause basic events in the Rev. 2 PRA model was potentially inadequate. No justification was provided for omitting a number of common cause component groups (check valves, batteries, etc.) typically found in other PRAs.

  For the LPI System, the only failure mode of interest for common cause consideration is failure of check valves to open on demand. A review of the industry database found only one Common Cause Failure (CCF) event of LPI check valves to open indicating a minor susceptibility to CCF. There were only a handful of events among Boiling Water Reactor (BWR) plants in their Residual Heat Removal systems, and all of these were check valves in the injection header side of the system. Given this information, it was concluded that the LPI injection header check valves 3LP-47 & 3LP-48 should be modeled for CCF in conjunction with the Core Flood check valves 3CF-12 and 3CF-14. Including these CCF events in the model does not impact the change in risk associated with the CT extension. A change in the CCF analysis affects both trains of LPI and not just one train.

Question 3:

Provide a discussion of Tier 2 considerations (RG 1.177) unless they're defined by compensatory measures.

Response:

Tier 2 provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change. As stated in the October 24, 2002 NRC submittal, the NRC SER found the change acceptable for Oconee due to the four listed compensatory measure that lower the risk impacts. As stated in the submittal, the four compensatory measures will be put in place prior to implementing the revision to the TS.

In addition, as also stated in the submittal, the avoidance of risk-significant plant configurations during the proposed LCO are controlled by procedures NSD 415, NSD 403, WPM 608, and WPM 609 which are used to address the Maintenance Rule requirement and the On-Line Maintenance Policy requirement to control the safety impact of combinations of equipment removed from service.

An update to the Oconee ORAM-Sentinel model is planned immediately following completion of the Rev. 3 Level 1 PRA analysis. However, the proposed changes are not expected to result in any significant changes to the current configuration risk management program. The existing program uses a blended approach of quantitative and qualitative evaluation of each configuration assessed. The Oconee ORAM-Sentinel model considers both internal and external initiating events with the exception of seismic events. Thus, the overall change in plant risk during maintenance activities is expected to be similar between Rev. 2 and Rev. 3 results.

Question 4:

Provide a statement that discusses the impact of the requested LPI change in Completion Time (CT) at power versus that at shutdown conditions since the LPI system is also used at shutdown conditions.

Response:

The risk impact of LPI (i.e., Decay Heat Removal (DHR) System) during shutdown is dependent on specific configurations and modes. In general, LPI is most risk significant for the prevention of core damage during shutdown conditions.

The importance of the LPI (DHR) system at shutdown is evidenced by the fact that MODE 5 and 6 Completion Times for system restoration are "Immediate" if the required loops are not OPERABLE. The requested LPI change will allow increased on-line preventative maintenance at power. Such maintenance is currently conducted during a shutdown. Additionally, the increased CT will allow longer corrective maintenance to be completed at power without requiring a plant shutdown. This change is expected to reduce the number of plant shutdowns. By maintaining the LPI (DHR) system at power, rather than during a shutdown when the DHR system is relied upon as the primary source of cooling, the risk during the shutdown is reduced.

Question 5:

State whether Revision 3 of the PRA finished or not, and if so, whether or not it's been used in the submittal. Are the conclusions of the analysis using Rev. 2 expected to be valid for Rev. 3?

Response:

Oconee PRA Revision 3 is not complete as of this date. It is expected to be completed later this year and therefore it has not been used in the quantitative part of the submittal. However, individual system models have been completed for Rev. 3 so insights can be gained from a review of this information.

Question 2 above discussed the changes to the Rev. 3 PRA based on the peer review and the impact of these changes on the LPI system. Additionally, the Rev. 3 LPI model itself included the following changes:

- Updated common cause failure probabilities
- Updated human reliability event data as needed
- Updated equipment reliability data as needed
- Implemented plant modifications on LPI system (deletion of valves and safety signals)
- Included LPI Pump C within the LPI System fault tree model

LOCA initiating event frequencies were revised for Rev. 3. LOCA sequences are important to the LPI TS change request:

PRA Rev. 3 utilizes data from NUREG/CR-5750 (1987-1995) for the generic frequency information.

A comparison of the LOCA initiating event frequencies used in the PRA is as follows:

Initiator	PRA Rev.2 (per year)	PRA Rev. 3 (per year)	PRA Rev. 3 (per year)
LL	3.38E-04	4.5E-06	4.5E-06 (total LL)
ML	3.38E-04	3.6E-05	9.4E-04 (total ML)
ML –Pzr. Safety	N/A	9.0E-04	
SL	1.44E-03	4.5E-04	1.6E-03 (total SL)
SL – Pzr. PORV	N/A	9.0E-04	
SL – RCP Seal	N/A	2.2E-04	

Utilization of the updated initiating event frequencies would increase the baseline CDF and the ICCDP values obtained.

System reliability data improved for the LPI system in Rev. 3. The table below compares the LPI pump failure probability values between Rev. 2 and Rev. 3:

Basic Event	PRA Rev.2	PRA Rev. 3
LPI Pump Fails to Start	3.71E-03	1.58E-03
LPI Pump Fails to Run	1.09E-03	5.18E-04
Common Cause: LPI Pumps Fail to Start	5.19E-04	1.00E-04

The revised estimates reflect both the additional plant specific data collected since the previous revision and a change to the source of generic data.

To estimate the impact of basic events important to the LPI analysis, a sensitivity study was run that changed the values in the Revision 2 cut set files (internal events only) to

Revision 3 values for the six basic events shown above. The results are as shown in the table below:

CDF values are per yr.	PRA Rev.2 (Internal Events Only)	Sensitivity Study (Internal Events Only)
Base Case CDF	2.61E-05	2.69E-05
CDF for 7 day CT (mean mnt. duration)	2.63E-05	2.71E-05
CDF for 7 day CT (full mnt. duration)	2.64E-05	2.73E-05
CDF Increase (mean mnt. duration)	2.0E-07	2.0E-07
CDF Increase (full mnt. duration)	3.0E-07	4.0E-07
Conditional CDF, 1 LPI train OOS	4.37E-05	4.65E-05
ICCDP for 7 day duration (unitless)	3.4E-07	3.8E-07

The results remain within the guidelines of RG 1.174 and 1.177.