April 29, 2005

LICENSEE: Carolina Power & Light Company

FACILITY: Brunswick Steam Electric Plant, Units 1 and 2

SUBJECT: SUMMARY OF TELECONFERENCE CONDUCTED ON MARCH 31, 2005, WITH CAROLINA POWER & LIGHT COMPANY (CP&L), TO DISCUSS THE SEVERE ACCIDENT MITIGATION ALTERNATIVES (SAMA) REQUESTS FOR ADDITIONAL INFORMATION (RAIs) FOR BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. MC4641 AND MC4642)

On March 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) staff and its contractors from Information Systems Laboratory conducted a teleconference with representatives from CP&L to discuss the SAMA RAIs for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP). The NRC staff formally sent the SAMA RAIs to CP&L by letter dated February 24, 2005, in support of the environmental review of the application for license renewal for BSEP. Enclosure 1 contains a listing of teleconference participants. Enclosure 2 is CP&L's draft response to item number 1h of the RAIs; this draft response was provided to the NRC by email dated March 30, 2005. The purpose of the call was to discuss RAI item 1h and CP&L's draft response. The meeting participants reached a shared understanding of RAI item 1h, and CP&L will adjust the draft response accordingly.

CP&L indicated that another teleconference will probably be needed when CP&L has finished drafting responses to the other RAI items. CP&L plans to submit the final responses to these RAIs by April 29, 2005.

/RA/

Richard L. Emch, Jr., Senior Project Manager
Environmental Section
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-324 and 50-325

Enclosures: As stated

cc w/enclosures: See next page

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DISTRIBUTION:

D. Matthews/F. Gillespie	R. Emch	A. Kugler	E. DiPaolo, SRI	P.T. Kuo
S. Hernandez-Quinones	S. Mitra	B. Mozafari	M. Lemoncelli, OGC	RLEP R/F
P. Frederickson, RII	J. Austin, RI	C. Guerrero	J. Fleming	R. Palla

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NAME	LJenkins	C. Guerrero	R. Emch	A. Kugler
DATE	04/27/05	04/27/05	04/27/05	04/29/05

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LIST OF PARTICIPANTS
TELECONFERENCE WITH CP&L REGARDING SAMAs

MARCH 31, 2005

Participants	**Affiliation**
Richard Emch	U.S. Nuclear Regulatory Commission (NRC)
Cristina Guerrero	NRC
Samuel Hernandez-Quinones	NRC
Robert Palla	NRC
Jason Fleming	NRC
Kimberly Green	Information Systems Laboratory (ISL), NRC's contractor
Bob Schmidt	ISL
Jan Kozyra	Carolina Power & Light Company (CP&L)
Dan Labelle	CP&L
Talmadge Clement	CP&L
Eric Browne	CP&L
John Caves	CP&L
Jeff Gabor	Erin Engineering, CP&L's contractor
Don MacLeod	Erin Engineering

Enclosure 1

Brunswick Steam Electric Plant
Units 1 and 2
cc:

Mr. Steven R. Carr
Associate General Counsel - Legal Department
Progress Energy Service Company, LLC
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Mr. David R. Sandifer, Chairperson
Brunswick County Board of Commissioners
Post Office Box 249
Bolivia, North Carolina 28422

Resident Inspector
U. S. Nuclear Regulatory Commission
8470 River Road
Southport, North Carolina 28461

Mr. John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street NW
Washington, DC 20037-1128

Ms. Beverly Hall, Section Chief
Radiation Protection Section,
Division of Natural Resources
N.C. Department of Environment
and Natural Resources
3825 Barrett Dr.
Raleigh, North Carolina 27609-7721

Mr. David H. Hinds
Plant General Manager
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461-0429

Public Service Commission
State of South Carolina
Post Office Drawer 11649
Columbia, South Carolina 29211

Ms. Margaret A. Force
Assistant Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
4326 Mail Service Center
Raleigh, North Carolina 27699-4326

Mr. T. P. Cleary
Director - Site Operations
Brunswick Steam Electric Plant
Progress Energy Carolinas, Inc.
Post Office Box 10429
Southport, North Carolina 28461-0429

Mr. Norman R. Holden, Mayor
City of Southport
201 East Moore Street
Southport, North Carolina 28461

Mr. Warren Lee
Emergency Management Director
New Hanover County Department of
Emergency Management
Post Office Box 1525
Wilmington, North Carolina 28402-1525

Mr. Chris L. Burton, Manager
Performance Evaluation and
Regulatory Affairs PEB 7
Progress Energy
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Mr. Edward T. O'Neill
Manager - Support Services
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

Mr. Allen K. Brittain
Superintendent, Security
Brunswick Steam Electric Plant
Progress Energy Carolinas, Inc.
P.O. Box 10429
Southport, North Carolina 28461-0429

cc:

Mr. Michael Heath
Brunswick Steam Electric Plant
P.O. Box 10429
Southport, North Carolina 28461-0429

Mr. Talmage B. Clements
Manager - License Renewal
Progress Energy
P. O. Box 1551
Raleigh, North Carolina 27602

Mr. Fred Emerson
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006-3708

Ms. Ilene Brown
NCUW
William Randall Library
601 S. College Rd.
Wilmington, NC 28403-5616

Pete Benjamin
Ecological Services Supervisor
U.S. Fish and Wildlife Service
Post Office Box 33726
Raleigh, NC 27636-3726

DRAFT

RAI Question Number: 1h

RAI Question:

It is stated that only 6 of 66 Level B facts and observations from the BWROG Peer Certification Review have been resolved in the version of the PSA used for the SAMA analysis. Provide additional information to substantiate the conclusion that no open issues would result in retention of a SAMA that was screened out based on the current PSA Model results.

RAI Response:

The SAMA analysis was based on the current model of record "MOR03" which existed at the time of the license renewal application. This PSA model was deemed acceptable by the peer review in that "all elements were consistently graded as sufficient for use in supporting risk informed decisions when combined with deterministic insights (i.e., a blended approach)". Resolution of the outstanding Level B facts and observations and update of MOR03 is still in progress. Much effort is being expended to resolve peer comments. Without the satisfactory completion of these changes, there is no qualified method for definitively answering the cost beneficial question raised by this RAI at this time. However, based on the nature of the modeling changes being considered and as discussed below, it is expected that there will be a small number of previously identified SAMAs that could change to cost beneficial or be further validated as cost beneficial.

The information included below addresses some of the major changes being made to address the Peer Review Certification comments, followed by an explanation of the impact on CDF and offsite consequences, and the potential to impact the overall SAMA conclusions.

The PSA model is being updated to address the remaining "B Level" facts and observations (F&Os) provided by BWROG Peer Certification team. The primary issues associated with the "B Level" facts and observations that are being addressed by the model update are as follows:

1. Need to address SRV (safety/relief valve) reclosure in DHR (decay heat removal) sequences where containment pressurizes.
2. Need to address NPSH issues in scenarios involving failure of suppression pool cooling and successful containment venting.
3. Need to address reactor building environmental conditions in scenarios where containment failure occurs prior to core damage.
4. Need to address potential conservatisms in the model dealing with common cause failure double counting, HVAC modeling for the diesel generator cells, failure of DC initiating events, modeling of CRD (control rod drive) initiating events, and including ARI (alternate rod injection) for ATWS events, and excluding manual shutdowns.
5. Need to address potentially non-conservative loss of offsite power initiating event data.

6. Need to refine the human error probability (HEP) estimates in the Human Reliability Analysis (HRA). The resolution of the HRA observations is expected to result in data enhancements by refining the bases used to define the HEPs and reducing the number of screening values used in the model.

The PSA model is being changed to more closely resemble the current NRC SDP (Significance Determination Process) event trees for BSEP associated with containment venting and late injection. The model is being changed to eliminate credit for late injection in sequences where all DHR has failed. The resultant changes to the model are intended to address SRV reclosure, NPSH issues, and the concerns about harsh environment in the reactor building after containment failure.

It has been confirmed that failure of CRD or loss of DC bus should be treated in the initiating event analyses. The selection of DC initiating events is being refined (eliminating some DC buses). Also, loss of 250 VDC is being added to the model to address potential common N/P bus failure. The CRD initiating events model is being retained (not excluded), but some refinements are being implemented in the logic to remove excess conservatism.

Several CCF (common cause failure) events for the support systems of the emergency diesel generators (EDG) are being removed from the model to more appropriately reflect component failure boundaries and to eliminate double counting. In addition, changes are being made to the success criteria for the EDG's HVAC to better depict its actual design bases and remove conservatisms (identified as overly conservative).

Updates are being performed to ATWS mitigation system reliability data (NUREG/CR-5500 Vol. 3), and logic changes are being incorporated to credit ARI.

The net result of all sequence modeling changes (e.g. SRV reclosure, NPSH, harsh environment) are expected to yield additional core damage sequences associated with loss of injection late, or complete loss of DHR (e.g. TQWZ). These sequences result in core damage and containment failure in time frames that exceed 30 hours from the event initiation. However, all of these TW (Class II) scenarios are being treated as intermediate time release categories based on inferred timing associated with the implementation of the Brunswick EALs (Emergency Action Levels) and the declaration of a General Emergency. It should be noted that modeling changes associated with the resolution of peer review facts and observations are expected to yield an increase in Class II sequences resulting in potential increase to release category H/I. This is a useful insight since it supports consideration of modifications that enhance the reliability of the DHR mitigation system. However, the following conservative modeling assumptions need to be considered in the evaluation of SAMAs:

1. All TW (Class II) sequences are assigned to the intermediate timing release categories (>6 hrs and <24 hrs) based on the inferred timing associated with the BSEP EALs. However, the supporting MAAP analysis (Ref. 1) indicates that core damage and containment failure is significantly delayed (> 24) in Class II sequences where CRD is available and would allow substantial time for operator actions.
2. The PSA model does not credit recovery of the condenser with the exception of LOOP scenarios (i.e. the probability of failing to restore offsite power is included in all LOOP sequences).

3. A conservative modeling error has been recently identified in the support systems for Hardened Wetwell Vent. The solenoid valves in the Nitrogen Backup system fail open on loss of power, and the power dependency should be removed from this model.
4. The Level 2 model uses screening values for some pre-initiators in sequences that currently do not significantly contribute to the release category profile.

The type of SAMA modifications that would help mitigate these Class II sequences are expected to involve improving the reliability of DHR and providing injection water to the containment. It should be noted that Phase II SAMA Number 36 addresses some of these issues and is already considered cost beneficial (see SAMA Appendix F.6.24).

Changes are being made to both the CRD and DC system initiating events. The changes to the CRD initiating event are expected to result in a significant reduction in the contribution for this initiator. The modeling changes to DC initiating events are expected to yield approximately the same absolute and relative contribution to the CDF initiating event risk profile. Since the contribution to CDF of these initiating events either is expected to reduce or remain practically unchanged, the consequences associated with these results would not be expected to result in the need to consider any additional SAMA modifications.

Changes to the EDG CCF and EDG HVAC success criteria are expected to result in a net reduction to the contribution of LOSP (loss of station offsite power) to the initiator distribution. These results would tend to reduce the contribution of early core damage due to loss of offsite power and would not be expected to result in the need to consider any additional SAMA modifications.

Similarly, the net changes to ATWS data are expected to result in a significant reduction in LERF since ATWS scenarios previously contributed approximately 75% of LERF. The significant reduction in LERF represents a significant decrease in the radiological release consequences, and thus would not be expected to result in the need to consider any additional SAMA modifications. The lower contribution to LERF may even result in a potential elimination of SAMAs that were retained in the analysis.