10 CFR 50.59(d)(2)

Progress Energy

Serial: RNP-RA/04-0042

APR 07 2004

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

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Ladies and Gentlemen:

Progress Energy Carolinas, Inc. (PEC), also known as Carolina Power and Light Company, submits the attached report in accordance with 10 CFR 50.59(d)(2), "Changes, Tests, and Experiments," for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The report provides a description of changes that were implemented pursuant to 10 CFR 50.59 between April 1, 2002, and April 1, 2004. A summary of the evaluation for each item is also included in the attached report.

If you have any questions concerning this matter, please contact me.

Sincerely,

C. T. Baucom
Supervisor – Licensing/Regulatory Programs

CAC/cac

Attachment

c: Mr. L. A. Reyes, NRC, Region II
NRC Resident Inspector, HBRSEP
Mr. C. P. Patel, NRC, NRR

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Summary of Changes, Tests, and Experiments for the
<u>H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2</u>

Evaluations performed in accordance with 10 CFR 50.59 for the time period of April 1, 2002, to April 1, 2004:

<u>Evaluation No. 03-0839</u>:

Description:

This evaluation pertains to the elimination of the containment hydrogen recombiner and hydrogen purge from design basis accident mitigation, as allowed by the revision to 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-water-cooled Power Reactors,” which was made effective on October 16, 2003.

10 CFR 50.44 no longer requires licensees with large dry Pressurized Water Reactor (PWR) containments to maintain recombiner or purge systems and procedures related to the control of combustible gas (hydrogen) levels in the containment following a design basis accident. Therefore, this 10 CFR 50.59 screening and evaluation are being applied to:

- The elimination of the hydrogen recombiner. This involves changes to the Updated Final Safety Analysis Report (UFSAR), the design basis document (DBD) for Post-Accident Heating Ventilation and Air Conditioning (HVAC) Systems, and other plant procedures related to obtaining, installing, and using the hydrogen recombiner.

- The elimination of the use of hydrogen purge for design basis accidents. Any aspects related to maintaining hydrogen concentrations less than design basis criteria may be eliminated. However, the Severe Accident Management guidelines/procedures retain some consideration for the possibility of using hydrogen purge. Therefore, procedural steps on the methods for performing a hydrogen purge continue to be included in such procedures.

- Some plant equipment, installed for the use of the recombiner, may be removed. If physical changes are made to the plant, an engineering change (EC) will be prepared. This evaluation may be referenced as part of the EC evaluation; however, completion of additional evaluations may be required to address other aspects of the physical change, such as the potential use of such equipment for functions other than hydrogen control. Changes to plant documents that are performed via a Configuration Management Update (CMU), such as a drawing change to eliminate reference to the hydrogen recombiner, are addressed by and can reference this evaluation.

The requirements for the hydrogen recombiner and for hydrogen purge are not contained in the HBRSEP, Unit No. 2, Technical Specifications. Therefore, NRC approval of a license amendment is not required prior to revising the design basis documents and procedures. Prior commitments that were made related to 10 CFR 50.44 are being eliminated as documented in this evaluation. NRC notification of the commitment changes, or NRC approval for elimination of these commitments is not required because the bases for the elimination of these commitments is fully addressed in the justification for the rule change and in the NRC Safety Evaluation Report (SER) for Improved Standard Technical Specifications Technical Specification Task Force generic change numbered TSTF-447.

Summary of Evaluation:

The activity only impacts procedures and equipment used after an accident has already occurred. The hydrogen recombiner is not installed in the plant and procedures for its installation and use would only be used following an accident. The procedural steps for a containment hydrogen purge would only be used following an accident. Therefore, deleting the requirements for these hydrogen control features from the procedures and from the design basis documents cannot increase the frequency of occurrence of a previously evaluated accident or create the possibility of a different type of accident.

The system, structure, or component (SSC) of concern is the containment. A malfunction of the containment would have significant safety consequences. In the UFSAR, the analysis of the containment shows that it will not experience a malfunction. It is assumed to leak at its design basis leak rate; however, containment integrity is not lost. Therefore, malfunction of the containment is not currently evaluated in the UFSAR.

The applicable accident is the Loss of Coolant Accident (LOCA), which is the only design basis accident (DBA) accident with sufficient hydrogen generation in the containment to warrant consideration of hydrogen control. Industry studies have demonstrated that containment integrity will not be challenged based on the ability of the containment to withstand the pressure spike from any hydrogen burn associated with the potential design basis levels of hydrogen in the containment. If the containment integrity is not challenged, then the consequences of the LOCA will not increase. In fact, the change could result in a reduction in the dose consequences of a LOCA due to the reduced likelihood of purging radioactivity from the containment to the environment for minimization of the hydrogen concentration. Additionally, the recirculation of radioactive containment air outside the containment through a recombiner system that could have leakage paths to the environment is eliminated.

The original intent of 10 CFR 50.44 was to ensure control of combustible gases, such that the potential for a flammable hydrogen concentration would be minimized and hence a pressure spike from a hydrogen burn would have minimal likelihood of causing a loss of containment integrity.

Regulatory Guide 1.7 specified models for calculating the buildup of hydrogen in the containment. The source of hydrogen from the zirconium-water reaction when fuel overheats was based on a very small fraction of the cladding (2%) reacting. This results in an initial containment hydrogen concentration in the range of 1%. The subsequent buildup of hydrogen with time then includes a source based on radiolysis of sump water, assuming 50% of the core inventory of iodine is in the sump. It is not possible to have that level of core iodine in the sump if minimal cladding has reacted. However, this yields a buildup from approximately 1% to a flammable concentration near 4% in a time frame from many days to many weeks. This time frame made the use of a recombiner and/or purging as viable methods for maintaining the hydrogen concentration less than flammable limits.

More realistically, if it is assumed that significant cladding reaction occurs, the hydrogen concentration would exceed 4% within hours of the LOCA. At Three Mile Island (TMI), a hydrogen burn occurred at approximately 8% concentration on the first day of the accident. For such a rapid buildup, a recombiner would be ineffective at preventing flammable concentrations. Purging would not be viable that early due to the dose consequences. Because of this potential (even though beyond design basis) for exceeding flammable concentrations early in the event, studies were performed on the ability of the containment to withstand a hydrogen burn. One such study is contained in the EPRI Nuclear Safety Analysis Center report NSAC/22, "Response of a TMI-2 Type Containment Structure to a 100% Hydrogen Burn," dated December 1981. This study states, "It was concluded that such a containment structure (large dry PWR containment) would be able to withstand the pressure from a burn of the hydrogen that would result from all of the zirconium in the core reacting with steam." Therefore, containment integrity, which includes the continued integrity of the containment penetrations, would not be challenged following a hydrogen burn and hence there would be no containment malfunction.

For the design basis LOCA, where minimal clad reaction occurs, the concentration will likely never reach flammable concentrations, or if such concentrations were approached, it would be sufficiently far into the future that means of minimizing the potential for exceeding flammable concentrations need not be included as a design basis requirement.

Based on these facts, the NRC concluded that 10 CFR 50.44 could be revised. The revision states that for large dry PWR containments there would be no requirement to assess the rate of buildup of hydrogen in the containment, and no requirement to provide controls, such as a recombiner or purging, to ensure a flammable mixture did not occur. (Note that the requirement to be able to measure the containment hydrogen concentration and to ensure good air mixing in the containment is being maintained. Additionally, the revised Regulatory Guide 1.7 still includes guidance to try and minimize the volume of metals in the containment, such as aluminum and zinc that could be a source of hydrogen in a LOCA environment. This evaluation is only addressing the elimination of the recombiner.)

In the September 16, 2003 Federal Register Notice of the rule change, the NRC states, "The final rule removes the existing definition of a design-basis LOCA hydrogen release and eliminates requirements for hydrogen control systems to mitigate such a release at currently-licensed nuclear power plants. The installation of recombiners and/or vent and purge systems previously required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design basis LOCA. The NRC finds that this hydrogen release is not risk significant. This finding is based on the Feasibility Study which found that the design basis LOCA hydrogen release did not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. The requirements for combustible gas control that were developed after the Three Mile Island Unit 2 accident were intended to minimize potential additional challenges to containment due to long term residual or radiolytically-generated hydrogen. The NRC found that containment loadings associated with long term hydrogen concentrations are no worse than those considered in the first 24 hours and therefore, are not risk significant. The NRC believes that the accumulation of combustible gases beyond 24 hours can be managed by licensee implementation of the severe accident management guidelines (SAMGs) or other ad hoc actions because of the long period of time available to take such action."

Based on the above, it is concluded that, for design basis accidents, the proposed changes will not result in a malfunction of the containment.

The changes described in this evaluation are only related to long term control of containment hydrogen concentrations. They have no impact on the fuel or Reactor Coolant System (RCS) fission product barriers. In regard to the containment barrier, the NRC has concluded that the elimination of these hydrogen control requirements does not have a risk significant impact on the containment barrier. No reanalysis of containment design basis limits, such as design pressure limits, is required because a hydrogen burn is not postulated to occur for a sufficiently long period of time following a design basis accident.

The UFSAR discusses analyses performed related to the buildup of hydrogen in the containment. These analyses were performed due to the requirements of 10 CFR 50.44. The revision to 10 CFR 50.44 eliminates the need for such analyses. Hence, this change is not revising any method of evaluation; it is simply eliminating the need for such an evaluation based on an NRC approved rule change.

The proposed changes revise station procedures and design basis documents to eliminate the requirements for a hydrogen recombiner and for hydrogen purge capabilities for design basis accidents. Such requirements had been included to meet 10 CFR 50.44 requirements to maintain control of post-accident containment hydrogen concentrations. The 10 CFR 50.44 rule change that became effective on October 16, 2003, eliminated these requirements for large dry PWR containments based on the conclusion that the levels of hydrogen possible following a design basis accident do not pose a challenge to containment integrity. Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

<u>Evaluation No. 03-0926</u>:

Description:

This evaluation pertains to changes made to the control room habitability toxic gas analysis. These evaluation changes resulted in revisions to the UFSAR, including updated information that describes current site and near-site toxic chemical inventories.

Summary of Evaluation:

These changes were determined to be an update of an existing analysis. Specifically, a new toxic gas analysis was performed. This calculation used new approved design inputs in determining the toxic gas concentrations in the control room following an accidental release of a toxic chemical. The parameters that were used in the analysis are the current design parameters. A new chemical survey of stationary and mobile sources was conducted. This survey provided current inputs to the toxic gas analysis. Postulated accidents include off-site mobile and stationary hazardous chemical spills, as well as on-site hazardous chemical spills. In general, a release rate is determined for each source identified. The dilution is modeled by atmospheric dispersion to determine the receptor concentrations. The modeling continues to use Gaussian dispersion and it was therefore concluded that a departure from the method of evaluation as described in the UFSAR had not occurred. The results of the analysis indicated that control room toxic gas concentrations continued to meet the applicable acceptance criteria.

Additionally, these analysis changes did not cause an increase in the frequency of occurrence of any accident or malfunction of an SSC important to safety, increase the consequences of any previously evaluated accident or malfunction, or create the possibility of an accident of a different type or a malfunction with a different result, because these changes were only related to the inputs and results of the affected analyses. As described above, the analysis changes did not result in a departure from a method of evaluation as described in the UFSAR. Also, these analysis changes did not directly or indirectly affect any design basis limit for a fission product barrier, as described in the UFSAR. Therefore, NRC review and approval of these changes prior to implementation was not required based on the conclusion that the proposed changes did not trigger any of the eight criteria listed in 10 CFR 50.59(c)(2).

Evaluation No. 03-1005:

Description:

This evaluation pertains to changes made to delete a restriction in Appendix B, "Additional Conditions," of the Operating License, which limited the cycle length to 504 Effective Full Power Days (EFPD). This restriction was added with the issuance of the power uprate license amendment (Amendment No. 196). The restriction was applied to ensure the current licensing basis radiological analyses remain bounding for operation at the uprated power. Additional analyses have been performed to justify operation for a full operating cycle.

Normally, a change to the Operating License would not require a 10 CFR 50.59 evaluation, as the change must be submitted to and approved by the NRC before implementation. However, Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," includes guidance on whether or not revised analyses require NRC review, based on the results of a 10 CFR 50.59 evaluation. Therefore, this 10 CFR

50.59 evaluation was performed to justify the scope of the license amendment request submittal to the NRC.

Summary of Evaluation:

The change to Appendix B of the Operating License deleted a restriction on Effective Full Power Days (EFPD) that was incorporated as a method to ensure the source term used for radiological dose analyses remained bounded by the dose analyses of record for operation at the approved uprated power level. The restriction was imposed solely for these post-accident radiological analyses assumptions. This change is not related to the probability or frequency of occurrence of an accident or the potential for a malfunction of any SSC, because this restriction was only related to post-accident analytical assumptions. Therefore, the change did not result in more than a minimal increase in the frequency of occurrence of an accident or a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Seven accident analyses that involve radiological consequences were addressed as part of the power uprate amendment. These accidents were the Fuel Handling Accident (FHA), the Loss of Coolant Accident (LOCA), Steam Generator Tube Rupture (SGTR), Main Steam Line Break (MSLB), Single Rod Cluster Control Assembly (RCCA) Withdrawal, Waste Gas System Failure, and Reactor Coolant Pump Shaft Seizure (Locked Rotor).

Based on evaluation of each of these accidents, it was concluded that operation at the uprated power, for cycles up to 567 EFPD, did not result in more than a minimal increase in the consequences of the current accident analyses. The change was unrelated to any malfunctions of an SSC. Therefore, the change did not impact any consequences related to malfunctions of an SSC. This change is not related to the initiation of an accident or malfunction of an SSC important to safety, because this restriction was only related to post-accident analytical assumptions. Therefore, the possibility of an accident of a different type or a malfunction with a different result was not created.

The NRC approved the removal of the 504 EFPD license condition by letter dated March 10, 2004. The enclosure to that letter states, "…the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed continued operation of HBRSEP2 beyond the current limitation of 504 EFPD. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 3.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 3.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed removal of the license condition limiting operation to 504 EFPD is acceptable with regard to the radiological consequences of postulated design-basis accidents."