HCGS 10CFR50.59 Evaluation No. HC 2008-215: "H-1-ZZ-MDC-1880, Revision 3"

50.59 REVIEW COVERSHEET FORM

Station/Unit(s): Hope Creek Generating Station

Activity/Document Number: 80096650

Title: Leakage Reduction Program Calculation, Revision 0

NOTE: For 50.59 Evaluations, information on this form will provide the basis for preparing the biennial summary report submitted to the NRC in accordance with the requirements of 10 CFR 50.59(d)(2).

Description of Activity:
(Provide a brief, concise description of what the proposed activity involves.)

The following activities are performed:
Increased the allowable Engineered Safety Feature (ESF) leak rate to 2.85 gpm
Updated the Main Steam Isolation Valve (MSIV) leakage release model
Increased the primary containment isolation valve (PCIV) maximum isolation time to 120 seconds, and
Revised the offsite and control room doses, the doses for the vital area missions, and the doses for areas requiring continuous occupancies to reflect the preceding activities.

Hope Creek Calculation H-1-ZZ-MDC-1880, Revision 3, evaluates the post-LOCA offsite and control room radiological impact of the following three changes:

1. Primary containment isolation valves (PCIVs) are proposed to remain open for 120 seconds during a LOCA. This change introduces a potential radioactive release path to the environment through the open drywell and suppression chamber purge exhaust valves.

2. Increase in the allowable ESF leak rate from 1.0 gpm to 2.85 gpm.

3. Update of the MSIV leakage release model to the current regulatory accepted model. The model changes include:

   a) Credited the elemental iodine removal by the containment wetted surface area,
   b) Revised the aerosol gravitational deposition in the MSIV lines beyond the outboard MSIVS to account for the finer aerosol particles by crediting a smaller aerosol removal rate than that in the current analysis,
   c) Modeled less elemental iodine removal in the main steam lines than that in the current analysis, and
   d) Redistributed the remaining MSIV leakage of 100 scfh in one intact main steam line instead of two steam lines in the current analysis.

The above changes result in decreases in the offsite radiological consequences, and an increase in the control room (CR) radiological consequences. The increase in the CR radiological consequence is both less than the regulatory allowable dose limit and can be defined as a minimal increase per the guidance in the 10 CFR 50.59 resource manual.

In addition to Calculation H-1-ZZ-MDC-1880, Revision 3, Technical Evaluation DCR # 80096650-0210, Rev 0 was originated, to determine the design functional impact on systems & components located downstream of the outboard PCIVs, which are expected to remain open for 120 seconds during a LOCA and exposed to peak LOCA pressure and temperature. The evaluation also assesses the impact of the increased maximum isolation time on the Primary Containment Integrated Leak Rate Test (PCILRT) Program, post-LOCA EQ temperature and doses, and structural integrity of system exposed to the post-LOCA peak pressure and temperature higher than the design condition.

Hope Creek Calculation H-1-ZZ-MDC-1923, Revision 2, evaluates the post-LOCA doses to area requiring continuous occupancy at the Technical Support Center (TSC), Guard House (GH), and Operational Support Center (OSC) due to changes in the post-LOCA release. The resulting TEDE doses are less than those calculated in the current analysis.

Hope Creek Calculation H-1-ZZ-MDC-1927, Revision 1, evaluates the post-LOCA mission doses to various vital areas due to changes in the post-LOCA release. The resulting TEDE dose rates are less than those calculated in the current analysis.

These changes hereafter are collectively called "proposed activity."

Reason for Activity:
(Discuss why the proposed activity is being performed.)

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The maximum PCIV isolation time was increased because compliance with the instantaneous closure times of 5 seconds listed in Hope Creek Technical Requirements Manual (TRM) Table 3.6.3-1 consistently became a difficult task for large bore PCIVs, mainly the drywell and suppression chamber purge supply and exhaust valves. The adverse impact of current instantaneous closure times is as follows:

- Large momentum associated with instantaneous closure time causes valve seat damage.
- Valve seat damage adversely impacts containment leak rate characteristics, long-term reliability, and PCILRT.
- Long-term valve seat damage results in an expensive valve replacement job.

The benefits of extended closure times are as follows:

- Permitted by the NRC in Regulatory Guide 1.183, Section 1.3.2, which allows the licensees to increase the PCIV maximum isolation time to 30 seconds without reanalyzing the design basis LOCA.
- Improves the long-term reliability of the PCIVs for the entire plant design life, including life extension of the plant
- Maintains the leak-tight containment pressure boundary and thereby improves the PCILRT results
- Provides operational flexibility
- Eliminates potential for expensive valve replacement costs
- Eliminates potential for a forced outage associated with the PCIV repairs

Per RG 1.183, Section 1.3.2, for the selected timing characteristics of the Alternative Source Term (AST) methodology, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be necessary if the modified elapsed time remains a fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase, which is 2 minutes or 120 seconds (RG 1.183, Table 4). This means that the NRC Staff allows the licensees to increase the PCIV maximum isolation time to 30 second (0.25 x 120 seconds = 30 seconds) without reanalyzing the design basis LOCA. For longer time delays, the regulatory guidance requires that the affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.

The PCIVs listed in the HC TRM Table 3.6.3-1 have been relocated from the HC Technical Specification by the HC operating license amendment No. 171. The PCIV maximum isolation times are proposed to increase up to 120 seconds due to the associated benefits listed above. The HC TRM maintains the maximum isolation times for the PCIVs, therefore the change to these times does not require an operating license amendment and NRC approval. The proposed changes can be adopted under the provisions of 10CFR50.59 guidance.

The review of the proposed change was performed using the applicable P&IDs to determine whether any of these open valves establishes a direct release path to the environment that bypasses the reactor building. The review indicates that the drywell purge exhaust (Penetration # 23, Isolation Valves GS-V024, V025, & V026) and suppression chamber purge exhaust (Penetration # 219, Isolation Valves GS-V027 & V028) could establish a direct release path to the environment during a LOCA. These purge exhaust isolation valves are proposed to remain open longer than 30 seconds; therefore, the evaluation of the radiological consequences became necessary for the 120 seconds closure time in H-1-ZZ-MDC-1880, Rev 3 to determine if the increases in the total dose consequences are less than minimal dose margins and if the total doses are less than the regulatory allowable limits to adopt the above changes for the current operating license under the provisions of 10CFR50.59 guidance.

A technical evaluation was performed to determine the design functional impact of systems & components located downstream of the outboard PCIVs, which are expected to remain open for 120 seconds during a LOCA and exposed to peak LOCA pressure and temperature. The evaluation also assesses the impact of the increased maximum isolation time on the PCILRT. The gas filled systems and components are exposed to the post-LOCA peak temperature and pressure, while the PCIVs are remaining open for 120 seconds during a LOCA. There is a potential impact on the systems and components design functional requirements due to the additional exposure to the post-LOCA conditions beyond the system design condition while the PCIVs remain opened.

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The allowable ESF leak rate was increased from 1.0 gpm to 2.85 gpm because plant maintenance has determined that meeting the 1.0 gpm leakage limit is difficult, a higher leak rate is believed to be bounding for future plant maintenance surveillances, and because the resultant total doses would not substantially increase above the current reported UFSAR doses.

The MSIV leakage release model was updated to be consistent with the current regulatory MSIV leakage model accepted and implemented to reflect the most recent NRC guidance as promulgated through NRC reviews and acceptance of the MSIV leakage model for the Peach Bottom Atomic Power Station. The adoption of the lately developed MSIV leakage model is appropriate to address the NRC concern about the lightly packed aerosols behavior in the main steam lines beyond the outboard MSIVs along with the reduction of the MSIV leakage based on the drywell pressure and temperature. The MSIV leakage model in the current analysis is extremely conservative, which unnecessarily expended the CR dose margin without having any prudent benefits. The newly adopted MSIV leakage is still conservative, complies with the NRC defense-in-depth philosophy, and is beneficial to the CR dose margin.

The radiological evaluation in H-1-ZZ-MDC-1880, Revision 3, determines that the total increase in dose consequences is minimal and that the total dose consequences are within the regulatory allowable limits. The technical evaluation concludes that the integrity of systems downstream of the outboard PCIVs are maintained without any adverse impact on their design functions and the increased maximum isolation time either totally eliminates or substantially reduces the large bore valve seat damage resulting in a leak tight pressure boundary during the PCILRT and following a LOCA.

Effect of Activity:
(Discuss how the activity impacts plant operations, design bases, or safety analyses described in the UFSAR.)

Post-LOCA exclusion area boundary (EAB), low population zone (LPZ) and control room (CR) doses are dependent on the activity released to the environment via different release paths. The inclusion of an additional bypass release path through open PCIVs, the increase in the allowable ESF leak rate to 2.85 gpm, and the updated MSIV leakage release model collectively reduced the offsite radiological consequences, and increased the control room radiological consequence. The total CR dose consequence and increase in the total CR dose consequence are both less than the regulatory allowable dose limit and can be defined as a minimal increase per the guidance in the 10 CFR 50.59 resource manual.

The increased maximum isolation time provides the operational flexibility and reduces the refueling outage critical time, and costs by either eliminating or minimizing valve seat damage and the need for repair or replacement of the large bore PCIVs having a virtually instantaneous closure time of 5 seconds. The containment pressure boundary can be tightly controlled during a LOCA to reduce the resulting dose consequences.

The increased ESF leak rate also provides operational flexibility by minimizing the likelihood of failed leak rate surveillance.

The structural integrity and design function of the systems downstream of the outboard PCIVs are not adversely impacted by their exposure to the post-LOCA peak pressure and temperature while these PCIVs remain open for 120 seconds during a LOCA.

The proposed change neither modifies the plant equipment design functions nor impacts the equipment reliabilities. It requires the revisions of valve testing procedures and HC TRM Table 3.6.3-1 for the increased closure time.

The post-LOCA dose rates to various vital access areas have been reduced with corresponding increases in occupancy times to perform the vital functions. The reduction in the post-LOCA vital access area dose rates and increases in occupancy times are not considered adverse because they are beneficial for the performance of the post-accident vital functions. Therefore these changes are screened out and are not subject to a 10 CFR 50.59 evaluation, even though the changes call for the LOCA safety analysis to be updated.

The Hope Creek UFSAR Change Notice No. HCN 08-028 identifies appropriate UFSAR changes.

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Summary of Conclusion for the Activity's 50.59 Review:
(Provide justification for the conclusion, including sufficient detail to recognize and understand the essential arguments leading to the conclusion. Provide more than a simple statement that a 50.59 Screening, 50.59 Evaluation, or a License Amendment Request, as applicable, is not required.)

The reanalysis of the radiological consequences of a LOCA to include the additional bypass release path through open PCIVs, which remain open for 120 seconds during a LOCA, to increase in the allowable ESF leak rate to 2.85 gpm, to update the MSIV leakage release model, and to revise the affected doses to various vital areas combined to result in decreases in the offsite radiological consequences, an increase in the control room radiological consequence, and an increase in allowable occupancy times to perform various vital functions. The total CR dose and increase in the CR dose consequence are both less than the regulatory allowable dose limit and can be defined as a minimal increase per the guidance in the 10 CFR 50.59 resource manual. The reanalysis does not:

- Adversely affect UFSAR described SSC design functions
- Adversely affect how UFSAR described SSC design functions are performed or controlled
- Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses
- Involve a test or experiment not described in the UFSAR
- Increase the frequency of occurrence of accidents
- Increase the likelihood of occurrence of malfunctions
- Increase the consequences of a malfunction
- Increase the possibility of an accident of a different type than is already analyzed in the UFSAR
- Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR
- Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered

Attachments:
Attach all 50.59 Review forms completed, as appropriate.
(NOTE: if both a Screening and Evaluation are completed, no Screening No. is required.)

Forms Attached: (Check all that apply.)

☒ **Applicability Review**				
☒ **50.59 Screening**	**50.59 Screening No.**	N/A	**Rev.**	N/A
☒ **50.59 Evaluation**	**50.59 Evaluation No.**	HC 08-215	**Rev.**	0

50.59 APPLICABILITY REVIEW FORM

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Address the questions below for all aspects of the Activity. If the answer is yes for any portion of the Activity, apply the identified process(es) to that portion of the Activity. Note that it is not unusual to have more than one process apply to a given Activity.

See Section 4 of the Resource Manual (RM) for additional guidance.

I.	Does the proposed Activity involve a change:		
	1. Technical Specifications or Operating License (10CFR50.90)?	☒ NO ☐ YES	See Section 4.2.1.1 of the RM
	2. Conditions of License Quality Assurance program (10CFR50.54(a))? Security Plan (10CFR50.54(p))? Emergency Plan (10CFR50.54(q))?	☒ NO ☐ YES ☒ NO ☐ YES ☒ NO ☐ YES	See Section 4.2.1.2 of the RM
	3. Codes and Standards IST Program Plan (10CFR50.55a(f))? ISI Program Plan (10CFR50.55a(g))?	☒ NO ☐ YES ☒ NO ☐ YES	See Section 4.2.1.3 of the RM
	4. ECCS Acceptance Criteria (10CFR50.46)?	☒ NO ☐ YES	See Section 4.2.1.4 of the RM
	5. Specific Exemptions (10CFR50.12)?	☒ NO ☐ YES	See Section 4.2.1.5 of the RM
	6. Radiation Protection Program (10CFR20)?	☒ NO ☐ YES	See Section 4.2.1.6 of the RM
	7. Fire Protection Program (applicable UFSAR or operating license condition)?	☒ NO ☐ YES	See Section 4.2.1.7 of the RM
	8. Programs controlled by the Operating License or the Technical Specifications (such as the ODCM).	☒ NO ☐ YES	See Section 4.2.1.7 of the RM
	9. Environmental Protection Program	☒ NO ☐ YES	See Section 4.2.1.7 of the RM
	10. Other programs controlled by other regulations.	☒ NO ☐ YES	See Section 4.2.1 of the RM
II.	Does the proposed Activity involve maintenance which restores SSCs to their original condition or involve a temporary alteration supporting maintenance that will be in effect during at-power operations for 90 days or less?	☒ NO ☐ YES	See Section 4.2.2 of the RM
III.	Does the proposed Activity involve a change to the:		
	1. UFSAR (including documents incorporated by reference) that is excluded from the requirement to perform a 50.59 Review by NEI 96-07 or NEI 98-03?	☒ NO ☐ YES	See Section 4.2.3 of the RM
	2. Managerial or administrative procedures governing the conduct of facility operations (subject to the control of 10CFR50, Appendix B)	☒ NO ☐ YES	See Section 4.2.4 of the RM
	3. Procedures for performing maintenance activities (subject to 10CFR50, Appendix B)?	☒ NO ☐ YES	See Section 4.2.4 of the RM
	4. Regulatory commitment not covered by another regulation based change process (see NEI 99-04)?	☒ NO ☐ YES	See Section 4.2.3/4.2.4 of the RM
IV.	Does the proposed Activity involve a change to the Independent Spent Fuel Storage Installation (ISFSI) (subject to control by 10 CFR 72.48)	☒ NO ☐ YES	See Section 4.2.6 of the RM

Check one of the following:

☐ If all aspects of the Activity are controlled by one or more of the above processes, then a 50.59 Screening is not required and the Activity may be implemented in accordance with its governing procedure.

☒ If any portion of the Activity is not controlled by one or more of the above processes, then process a 50.59 Screening for the portion not covered by any of the above processes. The remaining portion of the activity should be implemented in accordance with its governing procedure.

Signoff:

$\boxed{\text{50.59 Screener}}$/50.59 Evaluator: Gopal J. Patel Sign: [signature] Date: 11/15/2009

(Circle One) (Print name) (Signature)

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I. **50.59 Screening Questions** (Check correct response and provide separate written response providing the basis for the answer to each question)(See Section 5 of the Resource Manual (RM) for additional guidance):

1. Does the proposed Activity involve a change to an SSC that adversely affects an UFSAR described design function? (See Section 5.2.2.1 of the RM) ☒ YES ☐ NO

   The proposed activity of an increase in the allowable Engineered Safety Feature (ESF) leak rate and an update of the Main Steam Isolation Valve (MSIV) leakage release model do not involve a change to an SSC. The proposed activity of an additional release path associated with an increase in the primary containment isolation valve (PCIV) maximum isolation time to 120 seconds during a LOCA, and the design functional impacts on the systems, structures and components (SSCs) downstream of the open PCIVs, are evaluated in Reference II.2 for the SSC exposures to the post-LOCA containment peak pressure and temperature. The design pressures and temperatures of all systems downstream of the open PCIVs are less than the post-LOCA containment pressure and temperature, except for the primary containment instrument gas system (PCIGS) (Ref. II.2, Table 5), which has a design temperature that is less than the post-LOCA containment peak temperature. The structural integrity of PCIGS is further evaluated for the system exposure to a higher post-LOCA temperature imposing additional thermal expansion stress. The evaluation in Reference II.2, indicated that the total stress of various piping segments including the additional stress resulting from the post-LOCA temperature exposure is less than the allowable stress. Therefore, it is concluded that the structural integrity of the PCIGS will be maintained to perform intended normal and abnormal system functions.

   The SSC design functions described in the UFSAR are not adversely impacted by the proposed activity of the increased ESF leakage, updated MSIV leakage model, and opened PCIVs up to 120 seconds during a LOCA used in establishing the current design basis or used in the existing safety analysis. Although the open PCIVs establish the additional post-LOCA release path which contributes to offsite and control room doses that remain within the regulatory allowable limits (Ref. II.1, Sections 8.1 & 8.2), any increase in the dose exposure is considered adverse and requires further evaluation in the attached 50.59 Evaluation Form. The vital area mission dose rates are reduced and consequently the occupancy times have lengthened (Refs. II.8 & II.9). The reduction in the post-LOCA vital access area dose rates and increases in occupancy times are not considered adverse because they are beneficial for the performance of the post-accident vital functions. Therefore these changes are screened out and are not subject to a 10 CFR 50.59 evaluation, even though the changes call for the LOCA safety analysis to be updated. These changes are purely academic nature to demonstrate compliance with the NUREG-0737, Section II.B.2 shielding adequacy for performance of the potential vital functions. This information is historical in nature, never used for any post-accident action plan and does not adversely impact the design functions of SSC.

   In summary, the proposed activity alters the radiological design basis in the safety analysis, which needs to be evaluated in the attached 50.59 Evaluation Form.

2. Does the proposed Activity involve a change to a procedure that adversely affects how UFSAR described SSC design functions are performed or controlled? (See Section 5.2.2.2 of the RM) ☐ YES ☒ NO

   The proposed increase in the allowable ESF leak rate from 1.0 to 2.85 gpm provides the operational flexibility in the analyzed condition. The ESF leakage is postulated to determine the valid siting criteria for the Hope Creek site in compliance with the regulatory requirement (Ref. II.10). The increased ESF leak rate neither introduces a control mechanism nor alters the design functions related to the existing system configuration, and therefore does not adversely impact the manner in which the SSC design functions are performed or controlled.

   The proposed update of the MSIV leakage release model does not require a procedure change.

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The new limit of 120 seconds closure time for the PCIVs results in the revisions of I&C testing procedures, HC TRM Table 3.6.3-1 (Ref. II.1, Section 13.0), and UFSAR Table 6.2-16 to include the new closure time. As discussed in the response to Question 1, the increased closure time has no impact on SSC design functions. The increased closure time neither introduces a new control mechanism nor alters the design functions of the existing system configuration, and therefore does not adversely impact the manner in which the SSC design functions are performed or controlled.

In summary, the revisions of affected procedures do not adversely impact the UFSAR described SSC design function performance and control.

3. Does the proposed Activity involve an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses? (See Section 5.2.2.3 of the RM) ☐ YES ☒ NO

The proposed activity is analyzed in Reference II.1 using the AST methodology and TEDE dose criteria in accordance with Reg. Guide 1.183 (Ref. II.3) and ARCON96 atmospheric dispersion methodology in RG 1.194 (Ref. II.5). The NRC Staff approved these source term and atmospheric dispersion methodologies, and the TEDE dose criteria as HCGS licensing bases by issuance of operating license amendment 134 (Ref. II.4). The use of AST methodology, TEDE dose criteria, and ARCON96 atmospheric dispersion methodology to evaluate the radiological impact of the proposed activity is not an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses.

The proposed activity of an increase in the allowable ESF leak rate is consistent with the guidance of RG 1.183, Sections A5.1 through A5.6.

The proposed update of the MSIV leakage release model is also consistent with the guidance of RG 1.183, Sections A6.1 through A6.5 and is consistent with the most recent NRC guidance as promulgated through NRC reviews and acceptance of the MSIV leakage models for the Peach Bottom Atomic Power Station.

The proposed activity of an additional release path associated with an increase in the PCIV maximum isolation time to 120 seconds is consistent with the guidance of RG 1.183, Section 1.3.2.

In summary, the proposed activity is not an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses

4. Does the proposed Activity involve a test or experiment not described in the UFSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the UFSAR? (See Section 5.2.2.4 of the RM) ☐ YES ☒ NO

The proposed activity of an increase in the allowable ESF leak rate, an update of the MSIV leakage release model, and the additional release path associated with an increase in the PCIV maximum isolation time neither involve a test nor an experiment that is not described in the UFSAR.

The testing of the PCIVs will be performed in the same manner as before with the same applicable regulatory compliances with a newly established closure time without having any adverse effect on the plant safety and public health & safety.

In summary, the proposed activity does not involve any test or experiment not described in the

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UFSAR. The testing of the PCIVs will be performed in consistent with applicable testing procedures with a new closure time limit.

5. Does the proposed Activity require a change in the Technical Specifications or Operating License? (See Section 5.2.2.5 of the RM) ☐ YES ☒ NO

The ESF leak rate is not reflected in the Technical Specifications or the Operating License.

The MSIV leak rate is modeled consistent with the Technical Specification 3.6.1.2.c, "Primary Containment Leakage Limiting Condition For Operation".

The PCIVs listed in the Hope Creek Generating Station Technical Requirements Manual (HC TRM) Table 3.6.3-1 (Ref. 10A.6) have been relocated from the HC Technical Specification by the HC operating license amendment No. 171 (Ref. 10A.7) and their maximum isolation times are maintained in the HC TRM. Since the PCIV isolation times are controlled and maintained by the HC TRM outside the HC Technical Specifications, the change to isolation times does not constitute a change in the Technical Specifications or Operating License.

In summary, the change does not constitute a change in the Technical Specifications or Operating License.

II. List the documents (e.g., UFSAR, Technical Specifications, other licensing basis, technical, commitments, etc.) reviewed, including sections numbers where relevant information was found (if not identified in the response to each question).

1. Hope Creek Calculation No. H-1-ZZ-MDC-1880, Rev 3, Post-LOCA EAB, LPZ, and CR Doses
2. Hope Creek Technical Evaluation DCR # 80096650-0210, Revision 0, Technical Evaluation to Determine the post-LOCA Design Functional Impact on Systems & Components Located Downstream of Outboard Containment Isolation Valves which are Expected to Remain Open for 120 seconds at the Hope Creek Generating Station (HCGS)
3. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000
4. NRC Safety Evaluation Report, Hope Creek Generating Station – Issuance of Amendment No. 134 for Increase in Allowable MSIV Leakage Rate and Elimination of MSIV Sealing System
5. U.S. NRC Regulatory Guide 1.194, June 2003, "Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants."
6. Hope Creek Generating Station Technical Requirements Manual (HC TRM), Revision 1, Table 3.6.3-1, Primary Containment Isolation Valves
7. Hope Creek Operating License Amendment No. 171, RE: Relocate Component Lists For Primary Containment Isolation Valves From Technical Specifications (TAC No. MD3600)
8. Hope Creek Calculation No. H-1-ZZ-MDC-1923, Rev 2, Vital Area Mission Doses
9. Hope Creek Calculation No. H-1-ZZ-MDC-1927, Rev 1, Areas Requiring Continuous Occupancy
10. CFR 50.67, Accident Source Term

III. Select the appropriate conditions:

☐ If all questions are answered NO, then complete the 50.59 Screening and implement the Activity per the applicable governing procedure.

☒ If question 1, 2, 3, or 4 is answered YES and question 5 is answered NO, then a 50.59 Evaluation shall be performed.

☐ If questions 1, 2, 3, and 4 are answered NO and question 5 is answered YES, then a License Amendment is required

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prior to implementation of the Activity.

☐ If question 5 is answered YES for any portion of an Activity, then a License Amendment is required prior to implementation of that portion of the Activity. In addition, if question 1, 2, 3, or 4 is answered YES for the remaining portions of the Activity, then a 50.59 Evaluation shall be performed for the remaining portions of the Activity.

IV. Screening Signoffs:

50.59 Screener: Gopal J. Patel (Print name) Sign: [signature] (Signature) Date: 11/15/2009

50.59 Reviewer: AMITAVA GHOSE (Print name) Sign: [signature] (Signature) Date: 12/09/2009

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I. Complete the 50.59 Evaluation:

NOTES: Provide a separate written response providing the basis for the answer to each question below. The Resource Manual (RM) should be used to determine the content of each response (see Section 6.2 for additional guidance).

If the Screening indicated that only a change in method of evaluation exists, only Question 8 is required to be answered. If the Screening indicated that no change in method of evaluation exists, Question 8 does need not be answered.

1. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR? (See Section 6.2.1 of the RM) ☐ YES ☒ NO

The re-analysis of the radiological consequences due to an increase in the allowable Engineered Safety Feature (ESF) leak rate, an update of the Main Steam Isolation Valve (MSIV) leakage release model, and the additional release path associated with an increase in the primary containment isolation valve (PCIV) maximum isolation time does not introduce the possibility of a change in the frequency of an accident because these changes are not initiators of any accident and no new failure modes are introduced.

The design basis LOCA is categorized as a "Limiting Fault" in the HCGS UFSAR Section 15.6.5.1.2 (Ref. II.4). The LOCA is an event, which is not expected to take place, but is postulated because its consequences would include the potential for the release of significant amounts of radioactive material (Ref. II.1). Since the affected DBA is postulated to evaluate its dose consequences, the frequency of occurrence of the DBA is determined based on the Probabilistic Risk Assessment (PRA) and not dependent on the ESF leak rate, the MSIV leakage release model, or the additional release path introduced by the open PCIVs.

In summary, the proposed activity does not impact the frequency of occurrence of an accident previously evaluated in the UFSAR.

2. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR? (See Section 6.2.2 of the RM) ☐ YES ☒ NO

The re-analysis of radiological consequences does not introduce the possibility of a change in the likelihood of a malfunction because the re-analysis is not an initiator of any new malfunctions and no new failure modes are introduced.

The re-analysis of the radiological consequences does not introduce the possibility of a change in the likelihood of a malfunction because the design parameter values used in the analyses are consistent with the performance of the credited SSCs. The safety related function of the Filtration Recirculation and Ventilation System (FRVS) exhaust and Control Room Emergency Filtration (CREF) System is to mitigate the post-accident doses. The FRVS and CREF are technically credited in the revised analyses with the same filtration efficiencies and flow rates as those in the previous revision of the analyses. Therefore, the proposed change does not impose any additional challenges to their intended function and required performance, and do not increase the likelihood of occurrence of any malfunctions.

In summary, the proposed activity does not impact the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

3. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? (See Section 6.2.3 of the RM) ☐ YES ☒ NO

The following tables indicate that the changes in the revised radiological consequences of the design basis LOCA at the various receptor locations are less than minimal and the calculated total doses are less than allowable regulatory limits (Ref. II.1, Section 8.4). Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

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Design Basis Accident	Current Total Dose (rem) TEDE A	Proposed Total Dose (rem) TEDE B	Regulatory Dose Limit (rem) TEDE C	Proposed Dose Increase (rem) TEDE D=B-A	Minimum Dose Increase (rem) TEDE E=0.1(C-A)	RG Dose Limit (rem) TEDE F
Loss of Coolant Accident (LOCA)	H-1-ZZ-MDC-1880, Rev 2	H-1-ZZ-MDC-1880, Rev 3				
Control Room	4.16	4.17	5	0.01	0.084	5
Exclusion Area Boundary	3.10	1.43	25	-1.67	2.19	25
Low Population Zone	0.696	0.548	25	-0.15	2.43	25

B From H-1-ZZ-MDC-1880, Rev 3 (Ref. II.1)
C From 10 CFR 50.67 (Ref. II.3)
F From RG 1.183, Table 6 (Ref. II.2)

The inclusion of an additional bypass release path through open PCIVs, the increase in the allowable ESF leak rate to 2.85 gpm, and the update of the MSIV leakage release model combine to result in decreases in the offsite radiological consequences, and an increase in the control room radiological consequences. The post-LOCA proposed control room dose increase (in Column D) is less than the minimal dose increase regulatory limit (in Column E), and the post-LOCA total proposed control room dose (in Column B) is less than the allowable regulatory limit (in Column F). Therefore, the proposed activity does not result in more than a minimal increase in the consequences of the LOCA as previously evaluated in the UFSAR.

The vital area mission doses in calculation H-1-ZZ-MDC-1927, Rev 1 (Ref. II.11) determine the adequacy of the plant shielding to provide the required protection to an operator performing a post-accident vital function and occupancy based on the calculated maximum location specific dose rates. The post-LOCA dose rates and resulting occupancies in various vital areas are reported in the Safety Evaluation Report for Hope Creek Constant Pressure Power Uprate (Ref. II.20), Table 8-1. The vital area mission dose rates in the revised analysis are substantially reduced, which has allowed for increased occupancy times for the vital areas. Considering that the reduced radiation exposures are essentially the same and that they are calculated in a conservative manner, the resulting vital area mission doses and occupancies are acceptable without having any adverse impact on the current plant licensing bases.

Similarly, the doses for the areas requiring continuous occupancy (e.g., TSC, OSC, Guardhouse) in calculation H-1-ZZ-MDC-1923, Rev 2 (Ref. II.12) became less than the previously calculated doses. The post-LOCA doses for the areas requiring continuous occupancy are reported in Reference II.20, Tables 8-2 through 8-4. Considering that the reduced doses are essentially the same and that they are calculated in a conservative manner, the revised doses are acceptable without having any adverse impact on the current plant licensing bases.

In summary, there are only minimal increases in the total consequences of an accident and total dose consequences remain with the regulatory allowable limits.

4. Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR? (See Section 6.2.4 of the RM) ☐ YES ☒ NO

The re-analysis of radiological consequences does not introduce the possibility of a change in the consequences of a malfunction because the re-analysis is not an initiator of any new malfunctions and no new failure modes are introduced.

As discussed in Response to Question 2 above, the proposed change does not introduce any kind of malfunction of the safety related system credited in the revised analysis. The increased dose consequences neither credit any additional safety function nor involve any physical change to the SSC functions. Therefore, the proposed change does not create the possibility for a

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malfunction of an SSC important to safety previously evaluated in UFSAR and thereby it does not result in any related increase in the consequences.

In summary, there is no increase in the consequences of a malfunction.

5. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR? (See Section 6.2.5 of the RM) ☐ YES ☒ NO

The re-analysis of radiological consequences does not introduce the possibility of a new accident because the re-analysis is not an initiator of any accident and no new failure modes are introduced.

As discussed in Response to Question 1 above, the analyzed design basis LOCA is a hypothetical condition postulated because its consequences would include the potential for the release of significant amounts of radioactive material. The proposed change of an increase in the allowable ESF leak rate, an update of the MSIV leakage release model, and the additional release path associated with an increase in the PCIV maximum isolation time are not related to any mechanism or process that creates an accident. They simply represent the activity release paths contributing the dose consequences after the accident has already occurred. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

In summary, there is no increase in the possibility of an accident of a different type than is already analyzed in the UFSAR.

6. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR? (See Section 6.2.6 of the RM) ☐ YES ☒ NO

The proposed increase in the allowable ESF leak rate, an update of the MSIV leakage release model, and the additional release path associated with an increase in the PCIV maximum isolation time neither impact nor modify the SSC function. Consequently, the proposed activity neither involves any physical change to any SSCs nor modifies their existing important to safety functions. Therefore, the proposed activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR.

In summary, the proposed activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR.

7. Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? (See Section 6.2.7 of the RM) ☐ YES ☒ NO

There are three (3) fission product barriers namely the fuel cladding, reactor coolant system pressure boundary, and drywell (primary containment) pressure boundary. Two out of three fission product barriers - fuel cladding and reactor coolant system pressure boundary - are postulated to rupture during a typical large break LOCA. The fractions of fuel failure (core inventory) used in the reanalysis are the same as those modeled in the previous analysis. The PCIVs listed in the Hope Creek Generating Station Technical Requirements Manual (HC TRM) Table 3.6.3-1 (Ref. II.9) have been relocated from the HC Technical Specification Table 3.6.3-1 by the HC operating license amendment No. 171 (Ref. II.8) and their maximum isolation times are maintained in the HC TRM. The containment integrity is relaxed for 120 seconds when the drywell and suppression chamber purge exhaust PCIVs are postulated to remain open resulting in an additional bypass release path during a LOCA, which was analyzed. The instantaneous closure of PCIVs became necessary to maintain the containment integrity as a fission product barrier in the previous HC licensing basis based on the TID (Technical Information Document)-14844 source term, which postulates the instantaneous release of core inventory in the containment. To incorporate the NRC defense-in-depth philosophy to mitigate the dose consequences, the containment integrity was maintained by instantaneously closing those PCIVs, which establish a direct release path to the environment.

The NRC Staff approved the Alternative Source Term (AST), TEDE dose criteria, and ARCON96 atmospheric dispersion methodology (Ref. II.10) as the HCGS licensing bases by issuance of operating license amendments 134 and 146 (Refs. II.5 and II.18). In the AST (Ref. II.2, Table 1), only 5% of the core iodine and noble gas activity are expected to release into the

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containment 120 seconds (2 minutes) after onset of a LOCA. Therefore, the primary containment integrity as a fission product barrier is not required for the first 2 minutes after onset of a LOCA because no core fission products are released in the containment during this time period which would need containment confinement. The only activity available for release through the open containment barrier (PCIVs) is the reactor coolant specific activity, which is negligibly small in comparison to the core gap activity. The dose consequences from the open containment barrier via open PCIVs for 120 seconds before the onset of gap activity release are analyzed and added to other post-LOCA dose contributions. The total increases in dose consequences are less than applicable minimal dose margins as shown in Response to Question 3. Therefore, the relaxation of containment integrity as a fission product barrier while there is no fission product released in the containment is technically and legally acceptable per RG 1.183, Section 1.3.2. The re-analysis of other post-LOCA release paths credited the containment as a fission product barrier during and following the fission product release in the containment as described in the Hope Creek UFSAR and controlled by the Technical Specifications 6.8.4.f (Ref. II.6). Therefore, the fission product barrier limits described in the UFSAR are neither exceeded nor altered adversely.

In summary, the proposed activity neither exceeds nor alters the design basis limit for a fission product barrier as described in the UFSAR.

8. Does the proposed activity result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? (See Section 6.2.8 of the RM) ☐ YES ☒ NO

The revised analysis uses the AST methodology and TEDE dose criteria in accordance with Reg. Guide 1.183 (Ref. II.2) and ARCON96 atmospheric dispersion methodology in RG 1.194 (Ref. II.10). The NRC Staff approved these methodologies as HCGS licensing bases by issuance of operating license amendments 134 and 146 (Refs. II.5 and II.18). The aerosol deposition on the main steam lines surface areas used in the existing and revised analyses is consistent with the NRC approved guidance in AEB 98-03 (Ref. II.13). The aerosol deposition in the previous Revision 2 of the LOCA analysis was developed in a very conservative manner for the industry's first deposition model.

The previous deposition model had the following conservatisms in the analysis:

1. The MSIV leakage was assumed to be constant in both the MSIV failed steam line (150 scfh) and intact steam lines (50 scfh in two lines).
2. Forty (40) percentile aerosol settling velocity in piping between the RPV nozzle and Turbine Stop Valve (TSV).
3. Elemental iodine removal by the wetted surface area was not credited.

The previous deposition model was inconsistent with the latest regulatory development for aerosol deposition in the following manner:

1. The MSIV failed line boundary was not clearly defined. The NRC latest understanding and definition of the MSIV failed line is that it is the main steam line between the RPV and inboard MSIV and that aerosol and elemental iodine deposition and mixing in this failed line should not be credited. The MSIV failed line modeled in the previous analysis is inconsistent with the latest NRC definition of the MSIV failed line in that the previous analysis credited the aerosol and elemental removal and mixing in this segment of MSIV failed line without considering the volume of the pipe segment between the inboard and outboard MSIVs. In the revised LOCA analysis, the MSIV failed line boundary is clearly defined without aerosol and elemental iodine removal and without mixing in the MSIV failed line between the RPV nozzle and inboard MSIVS.
2. In the previous analysis, forty (40) percentile aerosol settling velocity was constantly used in both piping segments – RPV to outboard MSIV and outboard MSIV to TSV, which did not account for the newly developed NRC concept of lesser deposition of the lighter aerosol particle by gravitational deposition in piping beyond the outboard MSIV. The use of 40 percentile aerosol settling velocity is conservative in comparison to the NRC recommended aerosol settling velocity of 50 percentile for the heavier aerosol particles in the steam line between the RPV nozzle and outboard MSIV but its use for the entire release path from the RPV nozzle to TSV is non-conservative because it does not account for the lesser deposition of the lighter aerosol particles by gravitational deposition in the main steam line beyond the outboard MSIV.

The combined effect of both conservatisms and inconsistencies in the deposition models was such that the aerosol deposition model was very conservative in the previous analysis because previously the NRC unconditionally allowed the licensees:

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1. The reduction of the MSIV leakage based on the post-LOCA drywell pressure/temperature, and
2. The elemental iodine deposition on the wetted drywell surface areas.

Subsequent to the HCGS AST approval, the NRC Staff redefined the concept of an acceptable MSIV leakage model based on the research and experience gained with the AST. The revised LOCA analysis uses exactly the same MSIV leakage model that has been accepted by the NRC Staff in the following AST license amendments:

1. The reduction in the MSIV leak rates based on the post-LOCA drywell pressure/temperature in almost all BWR AST license amendments including, but not limited to, Dresden 2 and 3 (Ref. II.14), Quad Cities 1 and 2 (Ref. II.14), Peach Bottom 2 and 3 (Ref. II.15), Limerick 1 and 2 (Ref. II.16), and Clinton (Ref. II.17).
2. The adsorption of the elemental iodine by the drywell wetted surface area was credited in the AST license amendments for Dresden 2 and 3 (Ref. II.14), Quad Cities 1 and 2 (Ref. II.14), and Peach Bottom 2 and 3 (Ref. II.15).
3. The non-conservatism associated with the lighter aerosol particle removal by the gravitational deposition was a fairly new concern addressed by the NRC. Therefore, it is addressed only in the latest AST license amendments for Peach Bottom 2 and 3 (Ref. II.15), which already received its AST licensed amendment.

To make the revised analysis current and address the latest regulatory concern about the non-conservatism associated with the deposition of lighter aerosol particles, the revised analysis is analyzed exactly in the same manner as in the NRC approved AST license amendments as follows:

1. The MSIV leakage is postulated to release to the environment through two main steam lines instead of three lines in the previous analysis.
2. Each main steam release path is divided into two well-mixed volumes to be consistent with AEB 98-03.
3. A very well defined MSIV failed line boundary without crediting any deposition and mixing in the steam line between the RPV and inboard MSIV.
4. The fifty (50) percentile aerosol deposition velocity in the MSIV failed line between inboard and outboard MSIVs and intact steam line between the RPV nozzle and outboard MSIV is credited to justify that the heavier aerosol particles are subjected to a larger removal rate by gravitation and no aerosol deposition credited in the MSIV failed line between the RPV nozzle and inboard MSIV.
5. The thirty (30) percentile deposition velocity in the MSIV failed and intact steam lines beyond the outboard MSIVs to justify the reduction in the deposition of the lighter aerosol particle by gravitational deposition.
6. The elemental iodine removal by the drywell wetted surface area is credited in the analysis.
7. The lesser elemental iodine removal efficiency of 50% is used for 0-24 hrs in the main steam lines. No removal of elemental iodine is assumed after 24 hours.
8. No aerosol gravitation deposition of the lighter aerosol particles in the main steam lines is credited after 24 hours

Per PSE&G Procedure LS-AA-104-1000 (Ref. II.19), Section 3.4, rather than making a minor change to an existing method of evaluation, a licensee may adopt a completely new methodology without prior NRC approval provided the new method is approved by the NRC for the intended application. As discussed in Section 6.2.8, a new method is "approved by the NRC for the intended application" if it is approved for the type of analysis being conducted, the licensee satisfies applicable terms and conditions for its use, and the method is approved in an NRC SER or otherwise accepted by the NRC as part of a plant's licensing basis, e.g., the method is described in the plant's UFSAR or the NRC has accepted licensee commitments made in docketed licensing correspondence such as responses to NRC Generic Letters or Bulletins. Sections 3.4 and 6.2.8 note that the "conservative" and "essentially the same" criteria do not apply when evaluating the use of a new methodology approved by the NRC for the intended application. For example, the use of a new NRC-approved methodology may provide non-conservative results; however, that is acceptable as long as the methodology has previously been approved by the NRC for the intended application.

The Procedure LS-AA-104-1000 (Ref. II.19), Sections 3.4 and 6.2.8 describe two means in which one may depart from a method of evaluation described in the UFSAR. The second means is changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application. The second means is considered here appropriately and fully discussed in the above section.

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The revised analysis uses the same regulatory guidance described in the UFSAR as used in the current analysis. The revised analysis captures the excessive conservatism in the aerosol deposition model and updates the model to the latest requirements. The resulting dose margin is used to increase the ESF leakage and containment isolation time. Therefore, the reanalysis does not result in a departure from a method of evaluation that has been approved by NRC for the intended application without imposing any site-specific terms and conditions on other licensees. The change is not a departure from a method of evaluation because it is appropriate for the intended application, it complies with regulatory requirements, it is adopted in a conservative manner, and it has been approved by the NRC in other AST license amendments.

In summary, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

II. Identify references used to perform the evaluation (if not provided in the response to each question).

1. HCGS Calculation No. H-1-ZZ-MDC-1880, Rev 3, "Post-LOCA EAB, LPZ, and CR Doses"
2. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000
3. 10 CFR 50.67, "Accident Source Term."
4. HCGS UFSAR Section 15.6.5.1.2, Frequency Classification
5. NRC Safety Evaluation Report, Hope Creek Generating Station – Issuance of Amendment No. 134 for Increase in Allowable MSIV Leakage Rate and Elimination of MSIV Sealing System
6. HCGS Technical Specification 6.8.4.f, Primary Containment Leakage Rate Testing Program
7. Hope Creek Technical Specification Limiting Condition for Operation (LCO) 3/4.4.5, "Specific Activity"
8. Hope Creek Operating License Amendment No. 171, RE: Relocate Component Lists For Primary Containment Isolation Valves From Technical Specifications (TAC No. MD3600)
9. Hope Creek Generating Station Technical Requirements Manual (HC TRM), Revision 1, Table 3.6.3-1, Primary Containment Isolation Valves
10. U.S. NRC Regulatory Guide 1.194, June 2003, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments At Nuclear Power Plants."
11. Hope Creek Calculation No. H-1-ZZ-MDC-1927, Rev 1, Vital Area Mission Doses
12. Hope Creek Calculation No. H-1-ZZ-MDC-1923, Rev 2, Areas Requiring Continuous Occupancy
13. NRC Report AEB-98-03, "Assessment of Radiological Consequences For the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term
14. Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Issuance of Amendments RE: Adoption of Alternative Source Term Methodology (TAC NOs: MB6530, MB6531, MB6532, MB6533, MC8275, MC8276, MC8277, MC8278), September 11, 2006 (ADAMS Accession No. ML062070292)
15. Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Application of Alternative Source Term Methodology (TAC NOs: MD6806 and MD6807), September 5, 2006 (ADAMS Accession No. ML082320257)
16. Limerick Generating Station, Units 1 and 2 – Issuance of Amendments Re: Application of Alternative Source Term Methodology (TAC Nos. MC2295 and MC2296), August 23, 2006 (ADAMS Accession No. ML062210214)
17. Clinton Power Station, Unit 1 – Issuance of an Amendment RE: Application of Alternative Source Term Methodology (TAC No. MB8365), September 19, 2005 (ADAMS Accession No. ML052570461)
18. NRC letter to PSEG Nuclear dated April 15, 2003, "Hope Creek Generating Station – Issuance of Amendment 146 Re: Containment Requirements During Fuel Handling and Removal of Charcoal Filters (TAC No. MB5548)."
19. PSE&G Procedure LS-AA-104-1000, Rev 3, 50.59 Resource Manual.
20. GE NEDC-33076P, Rev 2, Safety Analysis Report for Hope Creek Constant Pressure Power Uprate

III. Based upon the results of this Evaluation: (Select one of the following)

☒ Implement the Activity per plant procedures without obtaining a License Amendment.

☐ Request and receive a License Amendment prior to implementation.

50.59 EVALUATION FORM

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IV. Signoffs:

50.59 Evaluator: Gopal J. Patel (Printed Name) _____ (Signature) Date: 11/15/2009

50.59 Reviewer: AMITAVA GHOSE (Printed Name) Amitava Ghose (Signature) Date: 12/09/2009

PORC Chairman: KENNETH M. KNAUSE (Printed Name) _____ (Signature) Date: 12/16/09

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Additional Proposed Changes to the HCGS Technical Specifications (Facility Operating License NPF-57)

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5.3.1	5-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. ~~and shall be limited to those assemblies which have been approved for use in BWRs~~ ***Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions***

A maximum of twelve GE14i Isotope Test Assemblies may be placed in non-limiting core regions, beginning with Reload 16 Cycle 17 core reload, with the purpose of obtaining surveillance data to verify that the GE14i cobalt Isotope Test Assemblies perform satisfactorily in service (prior to evaluating a future license amendment for use of these design features on a production basis). Each GE14i assembly contains a small number of Zircaloy-2 clad isotope rods containing Cobalt-59. Cobalt-59 targets will transition into Cobalt-60 isotope targets during cycle irradiation of the assemblies. Details of the GE14i assemblies are contained in GE-Hitachi report NEDC-33529P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Revision 0, dated December 2009,

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The absorber material has a nominal absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.
   2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 21,970 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.