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JAN 0 4 2002

PSEG
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LRN-01-0410
LCR H01-03

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001
Gentlemen:

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications for the Hope Creek Generating Station. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed amendment will add a Limiting Condition for Operation (LCO) for mechanical vacuum pump trip instrumentation. The need for this proposed change was identified during the reconstitution of the design basis analysis for the control rod drop accident. PSEG implemented administrative controls for the mechanical vacuum pump trip instrumentation in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety."

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1 to this letter. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 2. The supporting calculation is provided in Attachment 3.

PSEG requests approval of the proposed License Amendment by December 15, 2002 to be implemented within 60 days.

A001

Should you have any questions regarding this request, please contact Mr. Paul Duke at 856-339-1466.

Sincerely,

David F. Garchow
Vice President – Operations

Attachments (3)

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 1/4/02

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS FOR MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

1. DESCRIPTION

The proposed amendment would revise the Hope Creek Technical Specifications contained in Appendix A to the Operating License to add a Limiting Condition for Operation (LCO) for mechanical vacuum pump trip instrumentation.

2. PROPOSED CHANGE

The proposed changes to the Technical Specifications would add Technical Specification 3/4.3.10, "Mechanical Vacuum Pump Trip Instrumentation." The LCO would require that two channels of the main steam line radiation - high, high isolation function be capable of tripping the mechanical vacuum pumps. The trip function would be required to be OPERABLE when the plant is in OPERATIONAL CONDITIONS 1 or 2 with the mechanical vacuum pump in service and any main steam line not isolated. The Surveillance Requirement would provide appropriate requirements for CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST to ensure the mechanical vacuum pump trip instrumentation will perform its intended function.

The marked up Technical Specification pages are included in Attachment 2.

3. BACKGROUND

Two 50 percent capacity mechanical vacuum pumps are used during startup to establish a vacuum in the condenser. The mechanical vacuum pumps may also be used to maintain condenser vacuum following a plant shutdown/scram. The mechanical vacuum pumps are used when there is insufficient steam flow to operate the steam jet air ejectors. Plant procedures prohibit mechanical vacuum pump operation when reactor power exceeds 5%. If high radiation is detected in the main steam lines (detectors are located in the main steam tunnel between the outboard main steam isolation valves and the main steam stop valves) the pumps are automatically tripped, and the suction valves automatically close.

Amendment 53 to the Hope Creek Technical Specifications eliminated the requirements for scram and main steam line isolation valve (MSIV) closure associated with the main steam line radiation monitors. NRC approval of Amendment 53 was based in part on General Electric Licensing Topical Report NEDO-31400A (Reference 1) which demonstrates that removal of the automatic scram and MSIV closure functions does not cause the radiological release consequences of the bounding control rod drop accident (CRDA) to exceed acceptable dose limits. Eliminating these functions provides improved availability

of the main condenser for removal of decay heat and aids in eliminating inadvertent scrams.

The changes made in accordance with Technical Specification Amendment 53 did not affect the mechanical vacuum pump automatic trip and isolation function. The mechanical vacuum pump trip logic consists of two independent channels of the Main Steam Line Radiation - High, High function. The main steam line radiation monitoring system senses the gross release of fission products from the fuel and initiates appropriate actions to contain the released fission products. A trip of either channel is sufficient to result in a pump trip signal for both mechanical vacuum pumps.

The design for the mechanical vacuum pump trip function includes redundant safety related initiating logic up to the interface with the mechanical vacuum pump control circuits in the Bailey 862 Solid State Logic System. Downstream of the initiating logic, the trip function logic is neither safety-related nor single failure proof, similar to the design described in Carolina Power and Light Company's (CP&L's) license amendment request dated March 5, 1997 for Brunswick Unit Nos. 1 and 2. The NRC approved CP&L's request in a safety evaluation dated May 9, 1997 (TAC Nos. M98178 and M98179).

For the analysis of the case without automatic scram and MSIV closure, NEDO-31400A assumes the radiological release occurs via the main condenser offgas system. For a CRDA that occurs at low power without the offgas system operating, NEDO-31400A states that offsite dose impact would be equivalent to the case for a CRDA with automatic scram and MSIV closure.

As part of a reconstitution of the CRDA dose analysis, PSEG evaluated the consequences of a CRDA concurrent with mechanical vacuum pump operation. During this evaluation, PSEG concluded that automatic trip of the mechanical vacuum pump is required to ensure doses to the control room personnel do not exceed the limits specified in General Design Criterion 19 and Standard Review Plan Section 6.4.

10 CFR 50.36(c)(2)(ii), Criterion 3, requires that a Technical Specification LCO must be established for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Since the reconstituted design basis explicitly credits the automatic trip of the mechanical vacuum pump, this design feature needs to be included in the Technical Specification in accordance with 10 CFR 50.36(c)(2)(ii), Criterion 3. Technical Specification Table 3.3.2-1 Note (b) currently states that the Main Steam Line Radiation - High, High trip function also trips and isolates the mechanical vacuum pumps. However, there is no LCO for the mechanical vacuum pump automatic trip function.

4. TECHNICAL ANALYSIS

The proposed amendment would add LCO 3.3.10 and Surveillance Requirement 4.3.10 for the automatic trip of the mechanical vacuum pumps based on input from the main steam line radiation monitors. The LCO would require that two channels of the main steam line radiation - high, high isolation function be capable of tripping the mechanical vacuum pumps. The trip function would be required to be OPERABLE when the plant is in OPERATIONAL CONDITIONS 1 or 2 with the mechanical vacuum pump in service and any main steam line not isolated. The Surveillance Requirement would provide appropriate requirements for CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST to ensure the mechanical vacuum pump trip instrumentation will perform its intended function.

The need for this proposed change was identified during the reconstitution of the design basis analysis for the CRDA. A highly improbable combination of events is required for a CRDA to occur. These include the undetected failure of a control rod drive to control blade coupling; undetected sticking of the control blade in the upper part of the core; operator error in selecting and withdrawing an out of sequence control rod; and failure of the rod worth minimizer to block the out of sequence withdrawal. The rod worth minimizer functions to prevent withdrawal of an out of sequence control rod, minimizing the core reactivity transient during a rod drop accident.

The reconstituted analysis included an evaluation of the consequences of a CRDA concurrent with mechanical vacuum pump operation. The dose consequences for the CRDA were evaluated assuming the mechanical vacuum pump trips automatically due to either the Main Steam Line Radiation - High, High trip function or a loss of offsite power. With the mechanical vacuum pumps tripped automatically, doses to the control room operator do not exceed the limits specified in General Design Criterion 19 and Standard Review Plan Section 6.4.

The reconstituted CRDA dose analysis was performed before Hope Creek Technical Specification Amendment 134 was issued on October 3, 2001 for full implementation of an alternate source term (AST). Regulatory Guide 1.183 requires that the AST and TEDE criteria be incorporated into revisions to design basis radiological analysis performed after full implementation. Since this proposed change does not affect the analysis of radiological consequences for the CRDA, the analysis has not been revised. The calculation is included in Attachment 3.

The calculation was performed using the assumptions for a CRDA given in Section 15.4.9, Appendix A of the Standard Review Plan, NUREG-0800, (Reference 2). The Standard Review Plan requires that a loss of offsite power be assumed coincident with the CRDA. However, as discussed in NEDO-31400A (Reference 1), a loss of offsite power results in a loss of cooling water to

the condenser with eventual loss of condenser vacuum, resulting in automatic closure of the turbine stop and bypass valves, thus isolating the condenser from the reactor. The mechanical vacuum pumps also trip automatically upon a loss of offsite power. Therefore, even with a loss of offsite power, condenser leakage is not expected to exceed the 1% per day assumed in Standard Review Plan Section 15.4.9, Appendix A.

For release via an isolated condenser, site boundary doses were scaled directly from the values in Reference 1 and were confirmed to be less than the limits specified in Reference 2:

	Thyroid (Rem)	Whole Body (Rem)
Calculated EAB dose	0.35	0.025
SRP 15.4.9 Appendix A Limit	75	6

An assessment of control room doses at the control room air intake was performed using the TACT5 computer program in the HABIT computer code package. All doses are within the limits of Standard Review Plan 6.4 and General Design Criterion 19:

	Thyroid (Rem)	Whole Body (Rem)	Beta Skin (Rem)
Calculated dose at control room air intake	0.657	0.012	0.006
SRP 6.4 / GDC 19 Limit	30	5	30

The calculation also demonstrated that the doses to control room personnel due to the postulated CRDA were bounded by the analysis for the design basis loss of coolant accident (LOCA).

Credit for automatic tripping of the mechanical vacuum pump breakers is consistent with the assumptions in NEDO-31400A for the isolated condenser case (Scenario 1). It is also consistent with the analysis of the CRDA performed to support initial plant licensing documented in Section 15.4.9 of the NRC Safety Evaluation Report for Hope Creek, dated October, 1984 (NUREG-1048). This evaluation effectively did not credit MSIV closure since it assumed that 100% of the noble gases and 10% of the iodines released in the reactor vessel were transported to the condenser. Activity reaching the condenser was assumed to be release at a rate of 1% per day.

Automatic tripping of the mechanical vacuum pump breakers also causes the associated pump suction valves to close. However, automatic closure of the suction valves is not credited in the analysis in Attachment 3 or in the Updated Final Safety Analysis Report (UFSAR). Tripping the mechanical vacuum pumps is sufficient for mitigating the consequences of the postulated CRDA.

The proposed Technical Specifications for the mechanical vacuum pump trip instrumentation reflect the analysis discussed above. The mechanical vacuum pump trip is required to be OPERABLE in OPERATIONAL CONDITIONS 1 and 2 when any mechanical vacuum pump is in service (i.e., taking a suction on the main condenser) and any main steam line not isolated, to mitigate the consequences of a postulated CRDA. In OPERATIONAL CONDITION 3, 4 or 5 the consequences of a control rod drop are insignificant, and are not expected to result in any fuel damage or fission product releases. When the mechanical vacuum pump is not in service or the main steam lines are isolated, fission product releases via this pathway would not occur.

With one channel inoperable, but with mechanical vacuum pump trip capability maintained, the mechanical vacuum pump trip instrumentation is capable of performing the intended function. However, the reliability and redundancy of the mechanical vacuum pump trip instrumentation is reduced, such that a single failure in the remaining channel could result in the inability of the mechanical vacuum pump trip instrumentation to perform the intended function. Therefore, only a limited time (12 hours) is allowed to restore the inoperable channels to OPERABLE status. The 12 hour allowed outage time was shown to be acceptable in NEDC-30851P-A, "Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989. The exception to Specification 3.0.4 is consistent with the provisions of Technical Specification 3.3.2 for an inoperable isolation actuation instrumentation channel.

If the inoperable channel cannot be restored to OPERABLE status, or if mechanical vacuum pump trip capability is not maintained, the plant must be brought to an OPERATIONAL CONDITION or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least OPERATIONAL CONDITION 3 within 12 hours. Alternately, the associated mechanical vacuum pump(s) may be removed from service since this performs the intended function of the instrumentation. An additional option is provided to isolate the main steam lines which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser. The allowed completion time of 12 hours is reasonable, based on operating experience, to reach OPERATIONAL CONDITION 3 from full power conditions, or to remove the mechanical vacuum pump(s) from service, or to isolate the main steam lines, in an orderly manner and without challenging plant systems. The exception to Specification 3.0.4 is consistent with the provisions of Technical Specification 3.3.2 for multiple inoperable isolation actuation instrumentation channels.

An ACTION is also provided to allow that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated ACTIONs may be delayed for up to 6 hours provided mechanical vacuum pump trip capability is maintained. This allowance is based on the

reliability analysis in NEDC-30851P-A which demonstrates that the testing allowance does not significantly reduce the probability that the mechanical vacuum pumps will trip when necessary. In addition, the 6 hour test allowance is consistent with that previously approved for the main steam line radiation - high, high function in Technical Specification Amendment 70.

Appropriate CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST requirements are being added to ensure the mechanical vacuum pump trip instrumentation will perform its intended function. These requirements are also consistent with those previously approved for the main steam line radiation - high, high function in Technical Specification Amendment 70.

An Allowable Value is specified for the main steam line radiation-high, high trip function specified in the proposed Technical Specification. The nominal trip setpoint is specified in the setpoint calculations. The nominal setpoint is selected to ensure that the setpoint does not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable on the basis that the difference between the trip setpoint and the Allowable Value is an allowance for instrument drift.

The proposed change provides appropriate restrictions on plant operations consistent with the design basis analysis of the postulated control rod drop accident. In addition, the proposed change is consistent with NUREG-1433, Standard Technical Specifications, General Electric Plants, BWR/4, Revision 2, dated June, 2001.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment would add LCO 3.3.10 and Surveillance Requirement 4.3.10 for the automatic trip of the mechanical vacuum pumps based on input from the main steam line radiation monitors. The LCO would require that two channels of the main steam line radiation -

high, high isolation function be capable of tripping the mechanical vacuum pumps. The trip function would be required to be OPERABLE when the plant is in OPERATIONAL CONDITIONS 1 or 2 with the mechanical vacuum pump in service and any main steam line not isolated. Adding a requirement for the mechanical vacuum pump trip function does not affect any accident initiator. Automatic tripping of the mechanical vacuum pumps ensure that, following the postulated control rod drop accident, offsite doses at the exclusion area boundary are less than the limits specified in Standard Review Plan Section 15.4.9 Appendix A. Calculated doses to control room personnel are within the limits of Standard Review Plan 6.4 and General Design Criterion 19 of Appendix A to 10 CFR 50.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change adds Technical Specification requirements related to the automatic trip of the mechanical vacuum pumps based on input from the main steam line radiation monitors. It does not change the design function or operation of any systems, structures or components. Plant operation will not be affected by the proposed amendments and no new failure mechanisms, malfunctions or accident initiators will be created.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The safety related main steam line radiation monitors provide a reliable means to detect radioactivity resulting from a control rod drop accident and provide an automatic trip of the mechanical vacuum pumps to limit the release of radioactivity to the environment. Automatic tripping of the mechanical vacuum pumps ensure that, following the postulated control rod drop accident, offsite doses at the exclusion area boundary are less than the limits specified in Standard Review Plan Section 15.4.9 Appendix A. Calculated doses to control room personnel are within the limits of Standard Review Plan 6.4 and General Design Criterion 19 of Appendix A to 10 CFR 50.

Therefore, it the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50 Appendix A, General Design Criterion 19, "Control Room," requires that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

NUREG-0800, Standard Review Plan, Section 15.4.9, Appendix A, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, provides guidance to the NRC staff for review of the plant response to the postulated control rod drop accident, release of fission products, and calculation of whole body and thyroid doses.

The reconstituted design basis analysis of the radiological consequences associated with the postulated control rod drop accident is consistent with the guidance in Standard Review Plan, Section 15.4.9, Appendix A. Calculated doses are within the criteria of the Standard Review Plan and 10 CFR 50 Appendix A, General Design Criterion 19.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or a surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for

categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7. REFERENCES

1. General Electric Licensing Topical Report NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," dated October 1992.

2. NUREG-0800, Standard Review Plan, Section 15.4.9, Appendix A, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS

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INSTRUMENTATION

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.10 Two channels of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 with mechanical vacuum pump in service and any main steam line not isolated.

ACTION:

a. With one channel of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip inoperable, restore the channel to OPERABLE status within 12 hours. Otherwise, trip the mechanical vacuum pumps, or isolate the main steam lines or be in HOT SHUTDOWN within the next 12 hours.

   The provisions of Specification 3.0.4 are not applicable.

b. With mechanical vacuum pump trip capability not maintained:

   1. Trip the mechanical vacuum pumps within 12 hours; or

   2. Isolate the main steam lines within 12 hours; or

   3. Be in HOT SHUTDOWN within 12 hours.

   The provisions of Specification 3.0.4 are not applicable.

c. When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated ACTIONS may be delayed for up to 6 hours provided the mechanical vacuum pump trip capability is maintained.

SURVEILLANCE REQUIREMENTS

4.3.10 Each channel of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL CHECK at least once per 12 hours;

b. Performance of a CHANNEL FUNCTIONAL TEST at least once per 92 days;

c. Performance of a CHANNEL CALIBRATION at least once per 18 months. The Allowable Value shall be $\leq$ 3.6 x normal background; and

d. Performance of a LOGIC SYSTEM FUNCTIONAL TEST, including mechanical vacuum pump trip breaker actuation, at least once per 18 months.

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

BACKGROUND

The Mechanical Vacuum Pump Trip Instrumentation initiates a trip of the main condenser mechanical vacuum pump breaker following events in which main steam line radiation exceeds predetermined values. Tripping the mechanical vacuum pump limits the offsite and control room doses in the event of a control rod drop accident (CRDA). The trip logic consists of two independent channels of the Main Steam Line Radiation - High, High function. A trip of either channel is sufficient to result in a pump trip signal for both mechanical vacuum pumps.

APPLICABLE SAFETY ANALYSES

The Mechanical Vacuum Pump Trip Instrumentation is assumed in the safety analysis for the CRDA. The Mechanical Vacuum Pump Trip Instrumentation initiates a trip of the mechanical vacuum pump to limit offsite and control room doses resulting from fuel cladding failure in a CRDA (Ref. 1)

The mechanical vacuum pump trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of the mechanical vacuum pump trip is dependent on the OPERABILITY of the individual Main Steam Line Radiation - High, High instrumentation channels, which must have their setpoints within the specified Allowable Value of Surveillance Requirement 4.3.10.c. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the mechanical vacuum pump breakers.

APPLICABILITY

The mechanical vacuum pump trip is required to be OPERABLE in OPERATIONAL CONDITIONS 1 and 2 when any mechanical vacuum pump is in service (i.e., taking a suction on the main condenser) and any main steam line not isolated, to mitigate the consequences of a postulated CRDA. In this condition fission products released during a CRDA could be discharged directly to the environment. Therefore, the mechanical trip is necessary to assure conformance with the radiological evaluation of the CRDA. In OPERATIONAL CONDITION 3, 4 or 5 the consequences of a control rod drop are insignificant, and are not expected to result in any fuel damage or fission product releases. When the mechanical vacuum pump is not in service or the main steam lines are isolated, fission product releases via this pathway would not occur.

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION (continued)

ACTION a.

With one channel inoperable, but with mechanical vacuum pump trip capability maintained (refer to ACTION b Bases), the mechanical vacuum pump trip instrumentation is capable of performing the intended function. However, the reliability and redundancy of the mechanical vacuum pump trip instrumentation is reduced, such that a single failure in the remaining channel could result in the inability of the mechanical vacuum pump trip instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the low probability of extensive numbers of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of mechanical vacuum pump trip, 12 hours has been shown to be acceptable (Ref. 2) to permit restoration of an inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status, the plant must be brought to an OPERATIONAL CONDITION or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least OPERATIONAL CONDITION 3 within 12 hours. Alternately, the associated mechanical vacuum pump(s) may be removed from service since this performs the intended function of the instrumentation. An additional option is provided to isolate the main steam lines which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser.

The allowed completion time of 12 hours is reasonable, based on operating experience, to reach OPERATIONAL CONDITION 3 from full power conditions, or to remove the mechanical vacuum pump(s) from service, or to isolate the main steam lines, in an orderly manner and without challenging plant systems.

ACTION b.

ACTION b. is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels result in not maintaining mechanical vacuum pump trip capability. The mechanical vacuum pump trip capability is maintained when one channel is OPERABLE such that the Mechanical Vacuum Pump Trip Instrumentation will generate a trip signal from a valid Main Steam Line Radiation - High, High signal, and the mechanical vacuum pump breakers will open. This would require one channel to be OPERABLE, and the mechanical vacuum pump breakers to be OPERABLE. With mechanical vacuum pump trip capability not maintained, the plant must be brought to an OPERATIONAL CONDITION or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least OPERATIONAL CONDITION 3 within 12 hours. Alternately, the associated mechanical vacuum pump(s) may be removed from service since this performs the intended function of the instrumentation. An additional option is provided to isolate the main steam lines which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser.

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION (continued)

The allowed completion time of 12 hours is reasonable, based on operating experience, to reach OPERATIONAL CONDITION 3 from full power conditions, or to remove the mechanical vacuum pump(s) from service, or to isolate the main steam lines, in an orderly manner and without challenging plant systems.

ACTION c.

ACTION c. allows that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated ACTIONs may be delayed for up to 6 hours provided mechanical vacuum pump trip capability is maintained. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the required ACTIONs taken. This allowance is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the mechanical vacuum pump will trip when necessary.

Surveillance Requirement 4.3.10.a

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the required channels of this LCO.

Surveillance Requirement 4.3.10.b

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION (continued)

The frequency of 92 days is based on the reliability analysis of Reference 2.

Surveillance Requirement 4.3.10.c

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The 18 month frequency is conservative with respect to the assumption of the calibration interval in the determination of the magnitude of instrument drift in the setpoint analysis. For the purpose of this surveillance, normal background is the dose level experienced at 100% rated thermal power with hydrogen water chemistry at the maximum injection rate. The trip setpoint for the Main Steam Line Radiation - High, High trip function and requirements for setpoint adjustment are specified in Technical Specification 3.3.2.

Surveillance Requirement 4.3.10.d

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the mechanical vacuum pump breaker is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if the breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

The 18 month frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. UFSAR, Section 15.4.9.5.1.2

2. NEDC-30851P-A, "Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

CALCULATION H-1-CG-MDC-1795

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CALC NO.: H-1-CG-MDC-1795 REVISION: 2	CALCULATION COVER SHEET	Page 1 of 17

CALC. TITLE:	Control Rod Drop Accident - Analysis Reconstitution						
# SHTS (CALC):	17	# ATT / # SHTS:	3/4	# IDV/50.59 SHTS:	1/8	# TOTAL SHTS:	30

CHECK ONE:

☐ FINAL ☐ INTERIM (Proposed Plant Change) ☒ FINAL (Future Confirmation Req'd) ☐ VOID

SALEM OR HOPE CREEK: ☐ Q – LIST ☐ IMPORTANT TO SAFETY ☐ NON-SAFETY RELATED
HOPE CREEK ONLY: ☒Q ☐Qs ☐Qsh ☐F ☐R

☐ STATION PROCEDURES IMPACTED, IF SO CONTACT SYSTEM MANAGER
☐ CDs INCORPORATED (IF ANY):

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

Revised (see Order 70020574, Activity 0010) to incorporate a revised 10CFR50.59 Screening relating to Revision 1 of this calculation, which provided information relative to:

- Specific assumptions made (that is, the mechanical vacuum pumps are assumed to be tripped)
- *Evaluation against regulatory limits (that is, 10CFR100 and SRP Section 6.4 guidelines)*
- Explanation of any qualitative relationships to any other accidents described in the HCGS-UFSAR (that is, LOCA)

Moreover, the analysis is revised to correct the TACT5 input error identified in Notification 20035343.

PURPOSE:

To provide a reconstituted analysis of the radiation doses at the site boundary following a control rod drop accident (CRDA) in order to provide documentation for the radiological evaluations described in HCGS-UFSAR Section 15.4.9.5, "Radiological Consequences" including substantiation that main control room habitability for the CRDA is bounded by the analysis for the design basis loss-of-coolant accident (LOCA). Additionally, to evaluate the radiological consequences associated with a CRDA concurrent with mechanical vacuum pump operation.

CONCLUSIONS:

All the doses calculated are within acceptance criteria. That is,

a) for off-site doses (10CFR100 guidelines):

- 3.50E-1 rem < 6 rem whole-body
- 3.50E-1 rem < 75 rem thyroid

b) for control room doses (SRP Section 6.4 guidelines):

- < 1.23E-02 rem < 5 rem whole-body
- < 5.56E-03 rem < 30 rem beta skin
- < 6.57E-01 rem < 30 rem thyroid
- < 6.57E-01 rem thyroid post-CRDA < 0.896 rem beta skin post-LOCA

	Printed Name / Signature	Date
ORIGINATOR/COMPANY NAME:	J. Duffy/PSEG Nuclear	11/06/01
PEER REVIEWER/COMPANY NAME:	N/A	N/A
VERIFIER/COMPANY NAME:	R. Down/PSEG Nuclear	11/7/2001
PSEG SUPERVISOR APPROVAL:	G. Morrison	11/19/01

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ORIGINATOR,DATE	REV:	J. Duffy, 11/06/01	2				
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REVISION HISTORY

Revision	Description
0	Original Issue
1	Revised (see Order 70009023, Activity 0020) to provide information relative to: • Specific assumptions made (that is, the mechanical vacuum pumps are assumed to be tripped) • Evaluation against regulatory limits (that is, 10CFR100 and SRP Section 6.4 guidelines) • Explanation of any qualitative relationships to any other accidents described in the HCGS-UFSAR (that is, LOCA) Moreover, the analysis is revised to correct the TACT5 input error identified in Notification 20035343. Revision bars are not used due to the extent of the revision.
2	Revised (see Order 70020574, Activity 0010) to incorporate a revised 10CFR50.59 Screening relating to Revision 1 of this calculation.

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6	1	Attachment 11.3	-
7	1	1	0
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1.0 PURPOSE

To provide a reconstituted analysis of the radiation doses at the site boundary following a control rod drop accident (CRDA) in order to provide documentation for the radiological evaluations described in HCGS-UFSAR Section 15.4.9.5, "Radiological Consequences" including substantiation that main control room habitability for the CRDA is bounded by the analysis for the design basis loss-of-coolant accident (LOCA). Additionally, to evaluate the radiological consequences associated with a CRDA concurrent with mechanical vacuum pump operation.

Additionally, the radiological consequence analysis is revised (see Order 70009023, Activity 0020) to provide information relative to:

- Specific assumptions made (that is, the mechanical vacuum pumps are assumed to be tripped)
- Evaluation against regulatory limits (that is, 10CFR100 and SRP Section 6.4 guidelines)
- Explanation of any qualitative relationships to any other accidents described in the HCGS-UFSAR (that is, LOCA)

Moreover, the analysis is revised to correct the TACT5 input error identified in Notification 20035343.

2.0 BACKGROUND

Technical Specification Amendment 53 eliminated the main steam line radiation monitor (MSLRM) isolation of the main steam lines and automatic reactor shutdown features. The basis for the change was General Electric licensing topical report NEDO-31400A. The NRC approved the amendment on August 17, 1992. In so doing, NEDO-31400A became part of the HCGS design and licensing basis.

The GE topical report indicates that eliminating the scram and main steam isolation valve (MSIV) closure functions improved availability of the main condenser for decay heat removal and aids in eliminating inadvertent scrams. The GE topical report also indicated that other trip signals including mechanical vacuum pump remain functional.

HCGS-UFSAR Section 15.4.9 indicates that site boundary doses based on a Hope Creek specific atmospheric dispersion factor were calculated using the results presented in the GE topical report.

CR 990219176 is concerned with operating the mechanical vacuum pumps to evacuate the condenser during startup. Operating Procedure HC.OP-SO.CG-0001(Q) includes Precaution 3.1.2, which identifies that operation of the mechanical vacuum pumps while radioactive steam is being admitted to the main condenser will result in high radiation levels at the south plant vent. The procedure also includes Limitation 3.2.4, which calls for securing the mechanical vacuum pumps from service prior to reactor power exceeding 5%. Therefore, a control rod drop accident is postulated to occur when operating the mechanical vacuum

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pumps during startup when the MSIVs are open and before the steam jet air ejectors (SJAEs) are placed in service.

The confirmation for Order 70009023, Activity 0010, identifies that PSE&G submitted LCR H99-12, which requested NRC approval of an un-reviewed safety question related to a revised radiological analysis of the control rod drop accident (CRDA) for the Hope Creek Generating Station. As identified in associated Notification 20036248, an NRC technical reviewer questioned the basis for the following statement from the License Change Request (Attachment 1, page 5 of 5):

> ... the calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel.

Subsequent investigation showed the statement to have been inadequately substantiated in the LCR. Additionally, Notification 20035343 identified an incorrect computer input value for the initial I-131 condenser inventory (2700 Ci rather than 2770 Ci).

3.0 ANALYTICAL APPROACH

The model for calculating off-site whole-body and thyroid doses using conservative assumptions is identified in Standard Review Plan (SRP) Section 15.4.9, Appendix A.

The GE topical report indicates that GE calculated off-site doses using their proprietary CONACO3 computer program.

The TACT5 computer program in the HABIT computer code package is used by PSEG to calculate doses at the control room air intake location.

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4.0 DATA

Parameter	Value		Reference
Activity released from fuel (Ci)	I-131	2.77E5	GE internal memorandum DRR-89-07 dated 5/9/89 (a copy is attached as Attachment 1.1)
	I-132	4.04E5	
	I-133	5.79E5	
	I-134	6.37E5	
	I-135	5.46E5	
	Kr-83m	3.42E4	
	Kr-85m	7.34E4	
	Kr-85	3.29E3	
	Kr-87	1.41E5	
	Kr-88	2.00E5	
	Kr-89	2.48E5	
	Xe-131m	1.72E3	
	Xe-133m	2.51E4	
	Xe-133	6.03E5	
	Xe-135m	1.14E5	
	Xe-135	7.79E4	
	Xe-137	5.29E5	
	Xe-138	5.03E5	
Fission product transfer to main condenser	100% noble gas 10% iodine		SRP 15.4.9, Appendix A
Fraction of fission products airborne in the main condenser	100% noble gas 10% iodine		SRP 15.4.9, Appendix A
Condenser leak rate	1%/day		SRP 15.4.9, Appendix A
Condenser free volume	235,000 ft^3		HCCALC CG-0002
Mechanical Vacuum Pump flow rate	1900 cfm **Not used in the analysis**		HCDITS D3.6

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Parameter	Value	Reference
Number of operating mechanical vacuum pumps	2 **Not used in the analysis**	HCDITS D3.6
Steam Jet Air Ejector flow rate	75 scfm **Not used in the analysis**	HCDITS D3.6
Charcoal Holdup Time		
a) Normal operation	Krypton: 35.5 hr Xenon: 34.1 days	HCGS-UFSAR Table 15.4-6
b) Ambient operation	Krypton: 20.7 hr Xenon: 15.3 days	**NOTE: The reference is not a design basis document. The holdup times require future confirmation (see Notification 20035938).**
Site boundary χ/Q values (s/m^3)	0 - 2 hr 1.9E-4 2 - 4 hr 1.3E-4 4 - 8 hr 9.2E-5 8 - 24 hr 5.1E-5 1 - 4 day 2.5E-5 4 - 30 day 8.6E-6	H-1-ZZ-MDC-1820
Offsite breathing rate (m^3/s)	0 - 8 hr 3.47E-4 8 - 24 hr 1.75E-4 1 - 30 day 2.32E-4	RG 1.3
Control room air intake χ/Q values (s/m^3)	FRVS release: 0 - 8 hr 4.39E-5 8 - 24 hr 2.59E-5 1 - 4 day 1.64E-5 4 - 30 day 7.24E-6	HCCALC 19-0005

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Parameter	Value	Reference
Control room occupancy factors	0 - 24 hr 1 1 - 4 day 0.6 4 - 30 day 0.4	SRP 6.4
Control room breathing rate (m^3/s)	3.47E-4	Murphy-Campe paper (Ref. 8k)

5.0 ASSUMPTIONS

The expected MVP response following a CRDA is to be automatically tripped due to either loss of offsite power or a main steam radiation monitor signal. For example, procedure HC.OP-AB.ZZ-0203(Q) has a subsequent operator action to ensure the MVPs are out of service in response to main steam line high radiation. Therefore, the MVPs are assumed to be tripped (see Section 6.0, "Release with MVP operation", for further discussion).

6.0 DISCUSSION

HCGS-UFSAR Section 15.4.9 describes two transport pathways for the CRDA. One pathway considers holdup and decay in the Gaseous Waste Management System (GWMS). The other considers leakage of airborne activity from the condenser, if the GWMS is unavailable. These transport pathways correspond to the scenarios analyzed in the GE topical report. That is,

a) Scenario 1 - Analysis for CRDA with MSIV Closure, which corresponds to leakage of airborne activity from an isolated condenser

b) Scenario 2 - Analysis for CRDA without MSIV Closure, which corresponds to transport through the GWMS

HCGS-UFSAR Section 15.4.9.5 states that all of the iodine that enters the offgas treatment system is retained indefinitely and does not contribute to the off-site dose. The statement is consistent with GE's assumptions for the Scenario 2 analysis. Additionally, the GE topical report discussion for Scenario 2 indicates that if the event (that is, CRDA) occurs at low power without the SJAEs operating, the dose impact is bounded by Scenario 1. This is consistent with assuming that the mechanical vacuum pumps are tripped. Furthermore, concerning loss of offsite power, the GE topical report states that Scenario 2 will not result in a condenser leak rate exceeding the 1% per day assumption of SRP 15.4.9. This is also consistent with assuming that the mechanical vacuum pumps are tripped.

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HCGS-UFSAR Table 15.4-10 presents site boundary doses based on results presented in the GE topical report. These are:

	Whole-body dose (rem)	Thyroid dose (rem)
a) Release via GWMS at normal operating conditions (65°F)	2.03E-2	N/A
b) Release via GWMS at ambient operating conditions (77°F)	3.50E-1	N/A
c) Release via isolated condenser	2.50E-2	3.50E-1

The HCGS-UFSAR section states that the results are based on Hope Creek specific atmospheric dispersion factors.

Release via an isolated condenser

The GE topical report identifies that doses were calculated using an enveloping value of 2.5E-3 s/m^3 for the 2-hour χ/Q at the exclusion area boundary (that is, site boundary) for a ground v=level release. The GE topical report identifies the following doses:

- 4.3 rem (thyroid)
- 0.31 rem (whole-body)

The GE topical report states that doses for other χ/Q values may be scaled directly from these results. Using a χ/Q value of 1.9E-4 s/m^3 for the Hope Creek site boundary for 0 - 2 hours yields the following doses:

- (4.3 rem)(1.9E-4 s/m^3)/(2.5E-3 s/m^3) = 3.27E-1 rem (thyroid)
- (0.31 rem)(1.9E-4 s/m^3)/(2.5E-3 s/m^3) = 2.36E-2 rem (whole-body)

Therefore, the values shown in HCGS-UFSAR Table 15.4-10 are conservative.

Release via GWMS at normal operating conditions (65°F)

Figures 3 and 4 in the GE topical report present off-site doses due to krypton and xenon releases, respectively.

HCGS-UFSAR Table 15.4-6 shows the following holdup times for normal GWMS operation:

a) 35.5 hr for krypton

b) 34.1 days for xenon

The methodology for calculating charcoal holdup time is discussed in HCGS-UFSAR Section 11.3.2.1.2.1.

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The following site-boundary whole-body doses are obtained from GE topical report Figures 3 and 4 for the above holdup times and a χ/Q value of 3.0E-4s/m^3:

a) ≈ 9E-3 rem for krypton

b) ≈ 2.2E-2 rem for xenon

Therefore, the total whole-body dose is approximately 3.1E-2 rem.

9E-3 rem + 2.2E-2 rem = 3.1E-2 rem

Using a χ/Q value of 1.9E-4 s/m^3 for the Hope Creek site boundary yields 1.96E-2 rem.

(3.1E-2 rem)(1.9E-4 s/m^3)/(3.0E-4 s/m^3) = 1.96E-2 rem

Therefore, the value of 2.03E-2 rem that is shown in HCGS-UFSAR Table 15.6-10 is conservative.

Release via GWMS at ambient operating conditions (77°F)

Figures 3 and 4 in the GE topical report present off-site doses due to krypton and xenon releases, respectively.

HCGS-UFSAR Table 15.4-6 shows the following holdup times for normal GWMS operation:

a) 20.7 hr for krypton

b) 15.3 days for xenon

The methodology for calculating charcoal holdup time is discussed in HCGS-UFSAR Section 11.3.2.1.2.1.

The following site-boundary whole-body doses are obtained from GE topical report Figures 3 and 4 for the above holdup times and a site boundary χ/Q value of 3.0E-4s/m^3:

a) ≈ 2.5E-1 rem for krypton

b) ≈ 3E-1 rem for xenon

Therefore, the total whole-body dose is approximately 5.5E-1 rem.

2.5E-1 rem + 3E-1 rem = 5.5E-1 rem

Using a χ/Q value of 1.9E-4 s/m^3 for the Hope Creek site boundary yields 3.48E-1 rem.

(5.5E-1 rem)(1.9E-4 s/m^3)/(3.0E-4 s/m^3) = 3.48E-1 rem

Therefore, the whole-body dose value of 3.50E-1 rem that is shown in HCGS-UFSAR Table 15.6-10 is conservative.

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Release with MVP operation

As identified in the background discussion, a control rod drop accident is postulated to occur when operating the mechanical vacuum pumps during startup when the main steam isolation valves (MSIVs) are open and before the steam jet air ejectors (SJAEs) are placed in service.

SRP Section 15.4.9, Appendix A, identifies that a coincident loss of offsite power is assumed at the time of the accident. With loss of offsite power, if the mechanical vacuum pumps are running they will be tripped (see the mechanical vacuum pump response evaluation documented in Order 80031827, Operation 0010). GE Nuclear advises that mechanical vacuum pump trip is consistent with the SRP section assumptions concerning turbine and condenser integrity and turbine and condenser leakage at a rate of 1% per day (see Attachment 11.3).

Technical Specification Table 3.3.2-1 identifies that Main Steam Line Radiation - High, High trips and isolates the mechanical vacuum pumps. This is consistent with the statement provided in HCGS-UFSAR Section 15.9.6.5.3 that the main steam line radiation monitoring system will initiate the isolation of the reactor water sample valves and a mechanical vacuum pump trip on high high radiation in the main steam lines (also see the mechanical vacuum pump response evaluation documented in Order 80031827, Operation 0010). As stated above, GE Nuclear advises that mechanical vacuum pump trip is consistent with the SRP section assumptions concerning turbine and condenser integrity and turbine and condenser leakage at a rate of 1% per day. Therefore, condenser isolation is achieved even without loss of offsite power.

SER Section 10.4.2, Main Condenser Evacuation System, identifies that the NRC staff reviewed the Hope Creek system descriptions, piping and instrumentation diagrams, and design criteria for the components of the system and concluded that the system design was acceptable with respect to the control and monitoring of releases of radioactive materials to the environment. The mechanical vacuum pump trip is a feature of the system's radioactive material release control.

Control Room Doses

HCGS-UFSAR Section 15.4.9.5.1.4, "Main Control Room", states that main control room habitability for the CRDA is bounded by the analysis for the design basis loss-of-coolant accident (LOCA). The "analysis-of-record" for LOCA radiological consequences is Design Calculation H-1-ZZ-MDC-1822, which presents the following control room doses:

Whole body gamma dose (rem):	0.0367
Beta skin dose (rem):	0.896
Thyroid dose (rem):	0.524

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The corresponding SRP 6.4 guideline values are:

Whole body gamma dose (rem): 5

Beta skin dose (rem): 30

Thyroid dose (rem): 30

The control room doses expressed as percentages of the guideline values are:

Whole body gamma dose (%): 1

Beta skin dose (%): 3

Thyroid dose (%): 2

Therefore, the beta skin dose is limiting for control room habitability.

P&IDs M-07-1 and M-84-1 show that the MVPs discharge to the South Plant Vent (SPV). χ/Q values for releases from the FRVS exhaust located on the Reactor Building dome to the control room air intake are calculated in HCCALC 19-0005. Specific χ/Q values for releases from the SPV to the control room air intake were not calculated. However, the following results from Design Calculation H-1-ZZ-MDC-1879 show that the χ/Q values for releases from the FRVS exhaust to the control room air intake bound those for releases from the SPV to the control room air intake.

Time Interval (hr)	FRVS-to-CR χ/Q (s/m^3)	SPV-to-CR χ/Q (s/m^3)
0-2	1.25E-03	5.75E-04
2-8	8.09E-04	3.84E-04
8-24	3.04E-04	1.40E-04
24-96	2.10E-04	9.08E-05
96-720	1.59E-04	7.01E-05

Although the methodology used in Design Calculation H-1-ZZ-MDC-1879 differs from that used HCCALC 19-0005 (that is, ARCON96 vs. modified Halitsky), the H-1-ZZ-MDC-1879 results are sufficient to demonstrate that the FRVS χ/Q values bound the corresponding SPV χ/Q values. Therefore, using the FRVS χ/Q values from HCCALC 19-0005, which were calculated with our licensing-basis modified-Halitsky methodology, to model the release from the SPV is conservative. [Note: H-1-ZZ-MDC-1879 was performed using ARCON96 in support of a currently pending LCR. It is not, at this time, part of the HCGS licensing basis.]

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CALCULATION CONTINUATION SHEET

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CALC. NO.: H-1-CG-MDC-1795			REFERENCE: Order 70009023, Activity 0020				
ORIGINATOR,DATE	REV:	J. Duffy, 9/13/01	1				
~~REVIEWER~~/VERIFIER,DATE		K. Miller, 9/13/01					

With loss of offsite power, releases from the condenser would exfiltrate from the Turbine Building. Design Calculation H-1-ZZ-MDC-1879 computed the χ/Q values for releases from the Turbine Building from a location on the east side of the building (that is, an air intake louver) to the control room air intake. These values are also bounded by the FRVS χ/Q values.

Time Interval (hr)	FRVS-to-CR χ/Q (s/m^3)	Turbine Building-to-CR χ/Q (s/m^3)
0-2	1.25E-03	6.17E-04
2-8	8.09E-04	4.00E-04
8-24	3.04E-04	1.44E-04
24-96	2.10E-04	1.00E-04
96-720	1.59E-04	7.49E-05

Therefore, using the FRVS χ/Q values from HCCALC 19-0005 to model the release from the Turbine Building is conservative.

Doses at the control room air intake are conservatively estimated in the following manner using FRVS χ/Q values:

Thyroid dose:

(3.50E-1 rem)(24 hours/2 hours)(4.39E-5 s/m^3)/(1.9E-4 s/m^3) = 9.70E-1 rem

Whole-body dose:

(2.50E-2 rem)(24 hours/2 hours)(4.39E-5 s/m^3)/(1.9E-4 s/m^3) = 6.93E-2 rem

These doses are more than the corresponding post-LOCA doses in the control room documented in Design Calculation H-1-ZZ-MDC-1822 (that is, 0.524 rem and 0.0367 rem, respectively). However, doses in the control room would be even lower than those at the air intake due to dilution by uncontaminated air within the control room. Additionally, the results of recent control room inleakage tests indicate that the post-LOCA control room doses would be much higher than those documented in Design Calculation H-1-ZZ-MDC-1822 (see Notification 20073191 and Engineering Evaluation H-1-ZZ-MDC-1517) chiefly due to the less effective iodine removal by filtration that would be expected. Therefore, the post-CRDA control room doses are deemed to be bounded by post-LOCA control room doses based on engineering judgment.

A more accurate assessment of control room doses is performed using the TACT5 computer program in the HABIT computer code package. A modified version of computer file mlwricrp.30 (hconnew.dcf) contains dose conversion factors that are consistent with the isotopic data shown in HCGS-UFSAR Table 6.4-3. A copy of hconnew.dcf is included in Attachment 11.2, which contains the computer files used in this analysis, in subdirectory dcfs. The TACT5 computer output file is ccr--t5a.tab in subdirectory ccr. The following control room air intake results are obtained:

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- 1.23E-02 rem whole-body
- 5.56E-03 rem beta skin
- 6.57E-01 rem thyroid

All doses are within the acceptance criteria of SRP Section 6.4 (5 rem whole-body, 30 rem beta skin, and 30 rem thyroid) and are bounded by the limiting post-LOCA dose in the control room documented in Design Calculation H-1-ZZ-MDC-1822 (that is, 0.896 rem beta skin). The whole-body and beta skin doses are less than the corresponding post-LOCA doses in the control room documented in Design Calculation H-1-ZZ-MDC-1822 (that is, 0.0367 rem and 0.896 rem, respectively). However, the thyroid dose exceeds the corresponding post-LOCA thyroid dose shown in H-1-ZZ-MDC-1822 (that is, 0.524 rem). However, because of the control room inleakage issue identified above, the post-LOCA control room thyroid dose is deemed to bound the post-CRDA control room thyroid dose even without a control room emergency filtration system response, based on engineering judgment.

7.0 CONCLUSIONS/RECOMMENDATIONS

All the doses calculated are within acceptance criteria. That is,

c) for off-site doses (10CFR100 guidelines):

- 3.50E-1 rem < 6 rem whole-body
- 3.50E-1 rem < 75 rem thyroid

d) for control room doses (SRP Section 6.4 guidelines):

- < 1.23E-02 rem < 5 rem whole-body
- < 5.56E-03 rem < 30 rem beta skin
- < 6.57E-01 rem < 30 rem thyroid
- < 6.57E-01 rem thyroid post-CRDA < 0.896 rem beta skin post-LOCA

This reconstitution demonstrates that the values listed in HCGS-UFSAR Table 15.4-10 are accurate and conservative, and substantiates that the radiological consequences in the control room due to a CRDA are bounded by those for a DBA LOCA as described in HCGS-UFSAR Section 15.4.9.5.1.4.

8.0 REFERENCES

a) GE Report NEDO-31400A, dated October 1992, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor"

b) Standard Review Plan 15.4.9, Appendix A, Rev. 2, "Radiological Consequences of Control Rod Drop Accident (BWR)"

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c) Critical Software A-O-ZZ-MCS-0177, "Computer Codes for Evaluation of Control Room Habitability (HABIT)"

d) Letter NFS 96-370, "HCGS Design Basis LOCA Source Term Parameters"

e) GE internal memorandum DRR-89-07, dated May 5, 1989, "Activity Releases from the Fuel in CRDA Analyses for NEDO-31400"

f) HCDITS D3.6, Rev. 4, "Design, Installation, and Test Specification for Condenser Air Removal System for the Hope Creek Generating Station"

g) H-1-ZZ-MDC-1820, Rev. 0, "Offsite Accident Dispersion Factors"

h) Regulatory Guide 1.3, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors"

i) Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I"

j) HCCALC 19-0005, Rev. 2, "Control Room χ/Q Calculation and Diesel Exhaust Concentrations"

l) Standard Review Plan 6.4, Rev. 2, "Control Room Habitability System"

k) K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19," 13th AEC Air Cleaning Conference, August 1974

l) Procedure HC.OP-AB.ZZ-0203(Q), Rev. 5, "Main Steam Line High Radiation"

m) Procedure HC.OP-SO.CG-0001(R), Rev. 12, "Condenser Air Removal System Operation"

n) Design Calculation H-1-ZZ-MDC-1822, Rev. 0, "Loss of Coolant Accident Amendment 30 Model"

o) Design Calculation H-1-ZZ-MDC-1879, Rev. 0, "Control Room χ/Qs For South Plant Vent and Reactor Building Truck Bay"

9.0 FIGURES

None

10.0 TABLES

None

11.0 ATTACHMENTS

11.1 Copy of GE internal memorandum DRR-89-07, dated 5/9/89 (2 pages)

11.2 Zip 100MB disk with computer files (1 page):

mdc1795.doc (calculation file)

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dcfs subdirectory

- hconnew.dcf

ccr subdirectory

- conhab.dba
- ccr--cb.cnx
- ccr--cb.inp
- ccr--cb.run
- ccr--cb.spd
- ccr--cb.tab
- ccr--t5a.cnx
- ccr--t5a.inp
- ccr--t5a.nuc
- ccr--t5a.run
- ccr--t5a.tab
- ccr--t5b.cnx
- ccr--.dsg

11.3 Copy of telephone conversation record, dated 8/3/01, "Telephone Conversation Record Concerning Single Failure During A Postulated Control Rod Drop Accident", (1 page)

Attachment 11.1
H-1-CG-MDC-1795, Rev. 0
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DRR-89-07
5-09-89

cc: L.S. Burns
J.B. LaForce

TO: W. A. Zarbis

SUBJECT: Activity Releases from the Fuel in CRDA Analyses for NEDO-31400.

REFERENCE: 1. NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor", May 1987.

Fuel activity release data for the Control Rod Drop Accident analyses reported in Reference 1 was requested by Hope Creek. The attached Table provides activity releases from the fuel which are consistent with the condenser airborne activity inventories in Table 1 of Reference 1. The analysis was based on 850 failed fuel rods and a bounding power level of 0.12 Mw per rod.

D. R. Rogers

D. R. Rogers
Radiological and Shielding Analysis

Attachment 11.1
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ACTIVITY RELEASED FROM FUEL

Isotope	Activity (Ci)
I-131	2.77E+05
I-132	4.04E+05
I-133	5.79E+05
I-134	6.37E+05
I-135	5.46E+05
Kr-83m	3.42E+04
Kr-85m	7.34E+04
Kr-85	3.29E+03
Kr-87	1.41E+05
Kr-88	2.00E+05
Kr-89	2.48E+05
Xe-131m	1.72E+03
Xe-133m	2.51E+04
Xe-133	6.03E+05
Xe-135m	1.14E+05
Xe-135	7.79E+04
Xe-137	5.29E+05
Xe-138	5.03E+05

Zip 100MB disk with various electronic files

To: File

From: J. Duffy

Date: 8/3/01

Re: Telephone Conversation Record Concerning Single Failure During A Postulated Control Rod Drop Accident

R. Engel of GE Nuclear (telephone: 408-925-1016) called today in response to a message I left for Jim Leonard (telephone: 408-925-2164 concerning single failure during a postulated control rod drop accident. My concern was related to trip of the mechanical vacuum pumps (H1CG -1A-P-105 and H1CG -1B-P-105) and closure of the associated suction valves (H1CG -CG-HV-1979A and H1CG -CG-HV-1979B) following loss of offsite power or in response to high high radiation in the main steam lines.

Engel stated that the commitment to single failure was not well stated for this accident. However, he stated that it could be inferred that prior to the accident mechanical vacuum pumps could be running. He further stated that mechanical vacuum pump trip is consistent with Standard Review Plan Section 15.4.9, Appendix A, Radiological Consequences of Control Rod Drop Accident (BWR), assumptions concerning turbine and condenser integrity and turbine and condenser leakage at a rate of 1% per day. He further stated that closure of the associated valves is not needed to be consistent with the SRP assumptions (that is, mechanical vacuum pump trip is sufficient). Moreover, he stated that the mechanical vacuum pump trip is consistent with interpretation of IEEE 279.