Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUL 1 2002

Docket No. 50-423
B18678

RE: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 3
Response to a Second Request for Additional Information
Technical Specifications Change Request 3-2-00
Emergency Diesel Generator Allowed Outage Time

In a letter dated October 1, 2001,[(1)] Dominion Nuclear Connecticut, Inc. (DNC) requested changes to the Millstone Unit No. 3 Technical Specifications. The main purpose of the requested changes was to increase the allowed outage time for one emergency diesel generator from 72 hours to 14 days. In a letter dated May 13, 2002,[(2)] DNC submitted responses to seven (7) questions that were discussed during conference calls conducted on March 18 and April 4, 2002. On June 7, 2002,[(3)] a Request for Additional Information (RAI) was received via fax from the Nuclear Regulatory Commission which contains four (4) questions related to the aforementioned Technical Specifications Change Request.

Attachment 1 provides the DNC response to the June 7, 2002, RAI. This additional information provided in this letter will not affect the conclusions of the Safety Summary and Significant Hazards Consideration discussion in the DNC October 1, 2001, letter.

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(1) J. A. Price letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Technical Specifications Change Request 3-2-00, Emergency Diesel Generator Allowed Outage Time," dated October 1, 2001.

(2) J. A. Price letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Response to a Request for Additional Information, Technical Specifications Change Request 3-2-00, Emergency Diesel Generator Allowed Outage Time," dated May 13, 2002.

(3) V. Nerses (NRC) Memo to J. Clifford (NRC), "Millstone Nuclear Power Station, Unit 3, Facsimile Transmission, Draft - Request For Additional Information (RAI) to be discussed in an upcoming conference call (TAC No. MA3125)," dated June 7, 2002.

A001

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Sworn to and subscribed before me

this 1st day of July, 2002

Notary Public

My Commission Expires

WM. E. BROWN
NOTARY PUBLIC
MY COMMISSION EXPIRES MAR. 31, 2006

Attachment (1)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
NRC Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Attachment 1

Millstone Nuclear Power Station, Unit No. 3
Response to a Second Request for Additional Information
Technical Specifications Change Request 3-2-00
Emergency Diesel Generator Allowed Outage Time
Supplemental Information

Response to a Second Request for Additional Information
Technical Specifications Change Request 3-2-00
Emergency Diesel Generator Allowed Outage Time
Supplemental Information

In a letter dated October 1, 2001,[1] Dominion Nuclear Connecticut, Inc. (DNC) requested changes to the Millstone Unit No. 3 Technical Specifications. In a letter dated May 13, 2002,[2] DNC submitted responses to seven (7) questions that were discussed during conference calls conducted on May 18 and April 4, 2002. On June 7, 2002,[3] four (4) questions related to the emergency diesel generator (EDG) allowed outage time (AOT) Technical Specifications Change Request was received via fax from the NRC. The questions and associated responses are presented below:

Question 1

In the licensee's response to Question 3 (b), reference is made to both 1) Attachment 1, Sheet 4 of Work Management Procedure MP-20-WM-FAP02.1, "Conduct of On-Line Maintenance," and 2) Attachment 8 (MP-20-WM-FAP02.1). In order for the color code action levels presented in Attachment 1 and the color code risk matrix presented in Attachment 8 to provide meaningful information, the risk range associated with each color has to be provided. Specifically, the licensee is to provide for each color the range in instantaneous CDF associated with removing equipment from service.

Response

The risk color codes matrix presented in Attachment 8 to MP-20-WM-FAP02.1, "Conduct of On-Line Maintenance" are based on the calculated instantaneous core damage frequency (CDF) associated with removing equipment from service. The range for each color code is defined as follows:

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(3) V. Nerses (NRC) Memo to J. Clifford (NRC), "Millstone Nuclear Power Station, Unit 3, Facsimile Transmission, Draft - Request For Additional Information (RAI) to be discussed in an upcoming conference call (TAC No. MA3125)," dated June 7, 2002.

Green < 2E-04/yr
Yellow ≥ 2E-04/yr but < 4E-04/yr
Orange ≥ 4E-04/yr but <1E-03/yr
Red ≥ 1E-03/yr

Electric Power Research Institute (EPRI) Probabilistic Safety Assessment (PSA) Applications Guide (Final Report, TR-105396, August 1995) specifies that the risk levels should not exceed 1.0E-3/yr. Therefore, the threshold for entering the RED risk status is set to 1.0E-3/yr.

Question 2

In addition, since the matrix used at Millstone Unit No. 3 incorporates both the risk rate and accumulated risk and, hence, is potentially more meaningful than those typically used to manage risk, we need specific definitions of SCT and ACT (mathematical definitions, since they are calculated for each configuration and apply to multiple components simultaneously out of service) with some discussion of what they are intended to convey, in order to understand how the licensee intends to manage the risk associated with the proposed AOT extension using Attachments 1 and 8.

Response

"SCT" is defined as the scheduled configuration time. This parameter is derived from the plant's on-line work schedule and is based on the projected time we expect equipment to be removed from service to allow for test/repair.

"ACT" is defined as the allowed configuration time. This parameter is derived from the following equation:

$$ACT = \frac{[1E\text{-}06]}{R - BL}$$

where,

"R" = the instantaneous core damage frequency of a particular plant configuration, and

"BL" = the baseline core damage frequency with all equipment in service.

To ensure that the cumulative risk, expressed by the incremental core damage probability (ICDP), remains below 1.0E-6 (per NUMARC 93-01, Section 11), the SCT has to be less than the ACT which is calculated by the equipment-out-of-service (EOOS) computer program. By maintaining the "SCT" value less than the "ACT" value, we ensure "GREEN" risk status for the scheduled activities.

If the "SCT" value is greater than the "ACT" value but less than 10 times the "ACT" value, compensatory measures shall be established before entering the scheduled configuration. In such cases, the ICDP can be described by the following mathematical expression.

$$1.0E\text{-}6 \leq ICDP < 1.0E\text{-}5$$

If the "SCT" value is greater than 10 times the "ACT" value, the associated maintenance configuration shall not be entered voluntarily. In such cases, the risk status is "RED" and the ICDP is $\geq$ 1.0E-5.

Question 3

In its LAR, the licensee states that since the corrective action plan has not been implemented with regard to the WOG peer review (Sept. 1999), the findings were reviewed to identify those specifically applicable to the proposed EDG AOT extension and four sensitivity studies were initiated as compensation. In response to RAI Question 5, the licensee provided information from the peer review report on Objective and Approach, Scope, Process, Peer Review Grades, and Peer Review Team, as well as summary sheets on each of the eleven technical elements reviewed. Seven of the eleven elements (Containment Performance Analysis, Maintenance and Update Process, Accident Sequence Evaluation, Human Reliability Analysis, Dependency Analysis, Structural Response, and Quantification) were judged to be not adequate to support regulatory applications. The questions then are "What peer review findings prompted the sensitivity studies?" and "How do these studies compensate for the identified short comings (in the above seven technical elements) identified by the peer review?"

Response

The peer review findings that prompted the sensitivity studies are two fact and observation sheets that are considered directly related to the EDG AOT extension submittal. The observations and resolutions are as follows:

Fact and Observation 1: Millstone Unit No. 3 should keep abreast of the Westinghouse Owners Group (WOG) Reactor Coolant Pump (RCP) seal Loss of Coolant Accident (LOCA) modeling program regarding the interpretation of the NUREG/CR-4550 expert elicitation being used to establish a probabilistic RCP seal LOCA model.

Resolution 1: Since Millstone Unit No. 3 has not incorporated the latest WOG RCP seal leakage model, a sensitivity study was performed using the Rhodes model as the probabilistic RCP seal LOCA model. The Rhodes model was used since it has been recommended by the NRC for risk-informed licensing submittals.

Fact and Observation 2: While the 24-hour mission time is generally used, there are examples where it is bypassed. It is recommended that for seal LOCA sequences the mission time for successful mitigation be carried out at least until the leak rate is essentially eliminated via RCS depressurization, or at least 24 hours.

Resolution 2: Since Millstone Unit No. 3 has not incorporated this comment, a highly conservative bounding calculation of adding the small LOCA conditional core damage frequency to each seal LOCA core damage sequence was performed.

The sensitivity studies were not performed to compensate for the identified shortcomings in the peer review. The sensitivity studies were designed to identify significant contributors to core damage using worst case bounding assumptions. Compensatory measures were then developed based on the significant contributors identified. The actual impact on the Millstone Unit No. 3 core damage frequency, after the model is properly adjusted to account for these observations, is expected to be much less than the bounding estimates calculated by the sensitivity studies.

Although the peer review identified certain areas that need improvement, a systematic review of the seven technical elements was performed to justify that the current Millstone Unit No. 3 model is capable of supporting the EDG AOT risk-informed submittal. (Refer to Table 1)

Table 1
Disposition of Recommended Enhancements

Technical Element	**Recommended Enhancement**	**Affect on EDG AOT Submittal (LOOP/SBO Scenario)**
Accident Sequence Evaluation	Either reconstruct the technical bases of the accident sequence model from the Probabilistic Safety Study (PSS) or develop new bases from new thermal hydraulic analyses using MAAP and other appropriate engineering calculations. The updated documentation should be enhanced to address all the technical issues discussed above and in the Fact and Observation sheets for this Probabilistic Risk Assessment (PRA) element.	A technical basis was developed for the station blackout accident sequence using NUREG/CR-4550 as the basis for the timing and probability distribution of Reactor Coolant Pump (RCP) seal leakage events. A sensitivity study was also performed using the Rhodes model as the basis. The study concluded that the weather-related loss of offsite power event is a significant contributor to core damage. Consequently, extended EDG maintenance evolutions will not be scheduled when adverse or inclement weather conditions and/or unstable grid conditions are predicted or present.

Table 1
Disposition of Recommended Enhancements

Technical Element	Recommended Enhancement	Affect on EDG AOT Submittal (LOOP/SBO Scenario)
Human Reliability Analysis (HRA)	Follow the Millstone HRA guidance document. Include Type A errors, especially where multiple trains and/or systems may be affected. The methodology described in the guidance document should be implemented fully for all risk significant actions. The HRA screening values used in the model seem to be too low to be considered as screening values.	A detailed HRA evaluation for the operator action to start and align the station blackout (SBO) diesel had been completed prior to the peer review. The only other significant operator action for this scenario is recovery of a charging pump which has been included as part of the compensatory measures developed for this submittal. The PRA model does not consider a Type A or latent error that would disable the EDGs and the SBO diesel to be probabilistically significant.

Table 1
Disposition of Recommended Enhancements

Technical Element	Recommended Enhancement	Affect on EDG AOT Submittal (LOOP/SBO Scenario)
Dependency Analysis	The bases for addressing or not addressing spatial dependencies should be reviewed, brought up to date, and documented. Examples of spatial dependencies include High Energy Line Break (HELB) effects, flooding, spray effects.	The equipment required to mitigate a LOOP/SBO sequence is not susceptible to spatial effects. The EDGs, SBO diesel, and turbine-driven AFW pump are all housed in separate structures. Therefore, combinations of random equipment failures dominate the core damage sequences for the LOOP/SBO scenarios.
Structural Response	Update the Level 2 containment performance analysis, the Inter System Loss of Coolant Accident (ISLOCA) analysis, include pressurized thermal shock, and model Reactor Pressure Vessel (RPV) rupture events.	The probability of ISLOCA, RPV rupture, or pressurized thermal shock events occurring concurrently with LOOP/SBO sequences is negligible. Westinghouse performed a detailed Level-2 PRA analysis for Millstone Unit No. 3. The analysis included containment structural response as well as potential severe accident phenomena and their uncertainties. Since the state of knowledge of severe accident phenomenology did not advance substantially over the years since the Westinghouse study for Unit No. 3, the impact of not updating the Level 2 study on the EDG submittal is judged to be negligible.

Table 1
Disposition of Recommended Enhancements

Technical Element	Recommended Enhancement	Affect on EDG AOT Submittal (LOOP/SBO Scenario)
Quantification	Ensure PRA software versions have appropriate capabilities. Review dominant sequences for excessive conservatism and compare PRA results to those of similar plants to ensure consistency. Perform truncation studies and sensitivity studies to validate the model's results and perform at least a qualitative evaluation of uncertainties.	The latest versions of EPRI Risk and Reliability software tools were used. The dominant station blackout sequences were reviewed by an independent reviewer and approver. Significant contributors to core damage from station blackouts were identified and compensatory measures developed to reduce the core damage impact. Sensitivity studies were performed to identify significant core damage contributors due to station blackout. Insights from these studies were also used to develop compensatory measures designed to reduce core damage impact.

Table 1
Disposition of Recommended Enhancements

Technical Element	Recommended Enhancement	Affect on EDG AOT Submittal (LOOP/SBO Scenario)
Containment Performance Analysis	The Large Early Release Frequency (LERF) capability should be updated. Options include using NUREG/CR-6595 or a full scope update using MAAP 4.0. The plant damage states and Level 1/Level 2 interface will also have to be revised.	The thermal hydraulic results of MAAP 4.0 computer runs, which simulate SBO sequences, show that the earliest containment failure time is about 15 hours from accident initiation time. Therefore, the timing of radionuclide release to the environment does not represent an early release. Hence, it is expected that LERF results will not be impacted by the accident scenarios considered in the EDG AOT extension submittal.
Maintenance and Update Process	The process of model updates should include at least two key elements: a review of changes to the plant and operating experience and a rigorous review and validation of the results.	All plant modifications to Maintenance Rule risk significant systems are procedurally required to be reviewed by the PRA section. The PRA section reviews the shift manager logs monthly to calculate the core damage probability. The PRA analysis which forms the basis of the submittal was completed by an engineering calculation that included an independent reviewer and approver.

Question 4

According to the licensee, the LAR is supported by a PRA evaluation which utilized RG 1.177. The RG identifies a three-tiered approach. Tier 2 of the approach is to identify potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk-significant operational factors such as concurrent system or equipment testing were also to take place. The objective of this part of the evaluation is to ensure that appropriate restrictions on dominant risk-significant configurations associated with the change are in place. Although the licensee identifies 7 actions that are to be taken to minimize risk and claims that its CRMP provides assurances that high-risk configurations are precluded, high-risk configurations and appropriate restrictions were not identified. In response to Question 3 (a) the licensee begins to address Question 3 (b) by providing a list of risk significant equipment that is routinely removed from service for preventative maintenance during plant operation, instead of identifying high-risk configurations that are not to be entered during an EDG LCO AOT and saying what measures will be taken to ensure that the configurations do not occur. The CRMP, described by MP-20-WM-FAP02.1, does not contain a check list of identified equipment and appropriated restrictions, which is what we expect. Tier 2 is intended to show that the licensee has made an in-depth assessment of risks associated with the proposed TS change before planning maintenance, preventative or corrective.

Response

The original submittal (letter dated October 1, 2001) contained an evaluation of the Tier 2 requirement of Regulatory Guide 1.177. In that submittal, numerous plant configuration restraints and administrative controls were identified to limit plant risk while an EDG is out of service for an extended time period (i.e., up to 14 days). These additional requirements, applicable only when utilizing the proposed 14 day AOT, were specified in addition to use of the Configuration Risk Management Program (CRMP) which is required by 10 CFR 50.65(a)(4). Millstone Station complies with the CRMP through the implementation of 10 CFR 50.65(a)(4) requirements. The proposed additional requirements were designed to eliminate potential high risk configurations. Compliance with these additional requirements and utilization of the administrative controls already in place (Technical Specifications and CRMP) will reduce the plant risk of an EDG out of service for up to 14 days to an acceptable level.

The adequacy of the additional requirements was demonstrated in the October 1, 2001, submittal by calculating plant specific risk values. The calculated risk measures, which are below the RG 1.177 acceptance criteria of ICCDP < 5.0E-07 and ICLERP $\leq$ 5.0E-08, were based on the risk increase associated with the B train EDG (the EDG with the highest risk importance). The adequacy of also relying on administrative controls already in place (Technical Specifications and CRMP) was demonstrated by the response to Question 3 contained in the letter dated May 13, 2002.

The current Millstone Unit No. 3 Technical Specifications already provide administrative control over the removal of additional risk significant equipment when an EDG is out of service for up to 72 hours or for the proposed 14 days. The information presented in the response to Question 3 indicated that the calculated ACT was longer than the AOT specified by Technical Specifications for 20 of 23 equipment combinations. The ACTs for 2 of the 3 equipment combinations (46 hours for Train B Recirculation Spray System, 50 hours for Train B Motor Driven Auxiliary Feedwater Pump). Note that the calculated ACT of hours could vary, for the same plant configuration, if the PRA model is updated. The frequency of updating the plant PRA model is every refuel to reflect any risk-significant hardware changes, equipment reliability data, are approximately two-thirds of the Technical Specification AOT of 72 hours. Since work is not typically scheduled that will utilize more than one-half of the Technical Specification AOT (upper management approval is needed as specified in MP-20-WM-FAP02.1, Step 2.1.8), it is unlikely work affecting the operability of the Recirculation Spray System or the Auxiliary Feedwater System would be scheduled with an EDG out of service (either for the current 72 hour or for the proposed 14 day AOT). The most likely scenario for an EDG to be inoperable at the same time as the Recirculation Spray System or the Auxiliary Feedwater System would be the result of unexpected equipment failure. In that situation, the plant risk will be re-evaluated using the on-line risk assessment tool provided (e.g., equipment-out-of-service (EOOS)). For the two limiting configurations identified, sufficient time (approximately 2 days before the ACT would be exceeded) exists to repair the equipment. This should be sufficient for most equipment failures, especially since repair activities are typically performed around the clock for this type of risk significant safety equipment.

For the SBO DG, the proposed Technical Specification action requirement to restore the SBO DG to an available status within 72 hours is very close to the calculated ACT of 67 hours. The use of a 72 hour restoration time is consistent with the typical Technical Specification AOT. There is an insignificant adverse risk impact to plant safety associated with the 72 hour AOT instead of a 67 hour AOT. In addition the CRMP provides appropriate addition requirements if the ACT will be exceeded.

Providing an additional equipment checklist would be redundant to the real-time risk evaluation tools already utilized by the station and could lead to incorrect evaluation of the plant configuration. If an equipment checklist were to be developed, it would be based on an interpretation of the Millstone Unit No. 3 Technical Specifications. The process of developing this second level document could lead to confusion, and then this document may not accurately reflect subsequent changes to the Millstone Unit No. 3 Technical Specifications or the PRA model updates. To eliminate these potential errors, the Millstone Unit No. 3 Technical Specifications should be the base document used to control plant configurations, not a equipment checklist. The PRA evaluation tool EOOS is then used to enhance and support Technical Specification requirements. Hence, there would be no additional benefit associated with the development of an equipment checklist that is static in nature. Depending on the specific plant configuration, the allowed configuration time (ACT), which is calculated by EOOS, could be more conservative than the Technical Specification AOT. The EOOS computer program is utilized at the Millstone Station to implement the requirement of calculating the aggregate risk per 10 CFR 50.65(a)(4).