June 27, 2001

MEMORANDUM TO: James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Victor Nerses, Sr. Project Manager **/RA/**
Project Directorate I, Section 2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT 3, FACSIMILE TRANSMISSION, DRAFT REQUEST FOR ADDITIONAL INFORMATION (RAI) TO BE DISCUSSED IN AN UPCOMING CONFERENCE CALL (TAC NO. MA9740)

The enclosed draft RAI was transmitted by facsimile on June 27, 2001, to Mr. Ravi Joshi of Dominion Nuclear Connecticut, Inc. (DNC, licensee). After the clarification provided by the licensee in a previous conference call and subsequent staff evaluation, this draft RAI was transmitted to facilitate an additional conference call in order to further clarify the licensee's application dated July 25, 2000, regarding risk-informed inservice inspection program plan. Review of the RAI would allow DNC to determine and agree upon a schedule to respond to the RAI. This memorandum and the attachment do not convey a formal request for information or represent an NRC staff position.

Docket No. 50-423

Enclosure: Draft Request for Additional Information

June 27, 2001

MEMORANDUM TO: James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Victor Nerses, Sr. Project Manager **/RA/**
Project Directorate I, Section 2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT 3, FACSIMILE TRANSMISSION, DRAFT REQUEST FOR ADDITIONAL INFORMATION (RAI) TO BE DISCUSSED IN AN UPCOMING CONFERENCE CALL (TAC NO. MA9740)

The enclosed draft RAI was transmitted by facsimile on June 27, 2001, to Mr. Ravi Joshi of Dominion Nuclear Connecticut, Inc. (DNC, licensee). After the clarification provided by the licensee in a previous conference call and subsequent staff evaluation, this draft RAI was transmitted to facilitate an additional conference call in order to further clarify the licensee's application dated July 25, 2000, regarding risk-informed inservice inspection program plan. Review of the RAI would allow DNC to determine and agree upon a schedule to respond to the RAI. This memorandum and the attachment do not convey a formal request for information or represent an NRC staff position.

Docket No. 50-423

Enclosure: Draft Request for Additional Information

<u>DISTRIBUTION</u>:
PDI-2 Rdg PUBLIC
SDinsmore,SPSB SHou,EMCB VNerses

Accession No.: ML011780533
* see previous concurrence

OFC	PDI-2/PM	EMCB/SC*	SPSB/C*	
NAME	VNerses	ESullivan	MRubin	
DATE	6/27/01	6/26/01	6/27/01	/ /01

OFFICIAL RECORD COPY

DRAFT

REQUEST FOR ADDITIONAL INFORMATION
RISK-INFORMED INSERVICE INSPECTION (ISI) PROGRAM PLAN
MILLSTONE NUCLEAR POWER STATION, UNIT 3 (MP3)
DOMINION NUCLEAR CONNECTICUT, INC

The staff has completed its additional review of the MP3 request for relief from American Society of Mechanical Engineers (ASME) Section XI inservice inspection requirements. Based on this review, the staff has prepared requests for additional information (RAIs) which are provided below. These RAIs supplant and supercede the draft RAIs issued on May 10, 2001. The first twelve RAIs represent “generic” RAIs intended to improve the information content of the template submittals on risk.

1) One major step in the WCAP process is the identification of degradation mechanisms and the development of corresponding pipe failure frequencies. The requested Table 1 summarizes the qualitative results of this step by identifying the different degradation mechanisms, combinations of mechanisms, and the prevalence of the different mechanism. The calculated ranges in Table 1 summarize the quantitative results of the analysis. This information will illustrate how the degradation mechanism identification and failure frequency development step in the WCAP methodology was implemented, and provide an overview of the results generated. Please expand the current Table 3.4-1 to include the following information.

a) System	b) Degradation Mechanism/ Combination	c) Failure Probability range at 40 years with no ISI		d) Number of susceptible segments	e)Comments
		leak	disabling leak		

a) System: Each system included in the analysis.

b) Degradation Mechanism/Combination: Segment failure probabilities are characterized in the WCAP method by imposing all degradation mechanism in a segment (even if they occur at different welds) and the worst case operating conditions at the segment on a “representative” weld, and using the resulting failure probability for the segment. Please identify the dominant degradation mechanisms and combination of degradation mechanisms selected in each system. The reported mechanisms should cover all segments in the system. The table in the current submittal is not clear about which specific degradation mechanisms or combination of mechanisms are included in the leak estimates provided.

c) Failure Probability range at 40 years with no ISI: For each dominant degradation mechanism and combination of degradation mechanisms, please provide the range of estimates developed for the leak and disabling leak sizes as applicable. The table in the current submittal provided the range of leak estimates only.

d) Number of susceptible segments : Please identify the total number of segments susceptible to each dominant degradation mechanism and combination of degradation mechanisms.

e) Comments: The contents of this column are still being developed. It should provide further explanations and clarifications on the degradation mechanism and results as appropriate. Examples of items to be included are identification of which degradation mechanism are applied to socket welds and if a break calculation was needed to evaluate pipe whip constraints.

2) Another major step in the WCAP process is assignment of segments into safety significance categories based an integrated decision making process, and the selection of segments for inspection locations. The requested Table 3 summarizes the results of the safety significance categorization process as determined by the quantitative criteria, by the expert panel's deliberation on the medium safety significant segments, and by the expert panel's deliberations based on other considerations. The summarizing information requested in Table 3 will provide an overview of the distribution of the safety significance of the segments based on the quantitative results, and the final distribution based on the integrated decision making. Each segment has four Risk Reduction Worth (RRW) calculated, a Core Damage Frequency (CDF) with and without operator action, and a Large Early Release Frequency (LERF) with and without operator action. Please provide the following Table.

System	Number of segments with any RRW >1.005	Number of segments with any RRW between 1.005 and 1.001	Number of segments with any RRW between 1.005 and 1.001 placed in High Safety Significance (HSS)	Number of segments with all RRW < 1.001 selected for inspection

3) Another major step in the WCAP process is development of the consequences of segment ruptures. The WCAP methodology requires that a summary of the consequences be developed for each system and provided to the expert panel during their deliberations. Please provide this summary for each system. The summary will illustrate that the appropriate types of consequences (i.e., initiating events, mitigating system failure, and combinations) are included in the evaluation and will provide an overview of the results of the step.

4) Please add the statement that the sensitivity study to address uncertainty as described on page 125 of the WCAP was performed, and identify how many segments' RRW increased from below 1.001 to greater than or equal to 1.005. If the sensitivity study was not performed, provide a description and justification of any deviation.

5) Please state that the change in risk calculations were performed according to all the guidelines provided on page 213 of the WCAP or provide a description and justification of any deviation.

6) The quantitative change in risk results are adequately summarized in the current template tables 3-5 and 3-10. Please state that all four criteria for accepting the final selection of inspection locations provided on page 214 and 215 in WCAP-14572 Rev. 1-NP-A were applied. If all four criteria were not used, please provide a description and justification of the deviation. If comparison with any of the criteria indicated that “reevaluation” of the selected locations was needed, please identify the criteria that required the reevaluation and summarize the results of the reevaluation. If the results of the reevaluation can be found in the footnotes of Table 5-1, please refer to the footnotes.

7) Briefly describe the qualifications, experience, and training of the users of the Structural Reliability and Risk Assessment (SRRA) code on the capabilities and limitations of the code.

8) Intentionally left blank.

9) Please confirm that SRRA code was only used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1, page 15. For example, SRRA code should only be applied to standard piping geometry (circular piping geometry with uniform wall thickness). If the code was applied to any non-standard geometry, please describe how the SRRA inputs were developed.

10) Please describe any sensitivity studies performed to support the use of the SRRA code.

11) Intentionally left blank.

12) Please summarize the system design features and other physical characteristics of the plant as reflected in the risk evaluations that determined the location and the number of locations selected for inspection (This question was suggested for addition by Westinghouse at the May 22, 2001, public meeting).

13) Section 3.4 of your submittal states that, “Generally, the SRRA code was used to estimate where the possible ranges of failure probability would fall. The final probability selected was determined by team members using the relevant information." Page 83 of the Topical states that for Westinghouse Owners Group (WOG) plant application, “(SRRA) tools were used to estimate the failure probabilities for the piping segment”. Page 6 and 7 of the related safety evaluation also state that the failure probability estimate, “is subsequently used to represent the failure probability of the weld.” Please explain how your method comports with the approved Topical and the Safety Evaluation. Please also provide an example of the maximum range provided to the expert panel from which to select a value.

14) In the staff's IPE data base (a data base that includes the results of all the original submitted versions of the Ibes) the MP3 Conditional Core Damage Probability (CCDP) for large Loss of Coolant Accident (LOCA) (>6") is 2.07E-2, Medium LOCA (between 2" and 6") is 1.69E-2, Small LOCA (between 3/8 and 2") is 4.00E-4.

> a) What CCDPs and Conditional Large Early Release Probabilities (CLERPs) do you currently have for these LOCA sizes? If the CCDPs and/or CLERPs are location dependent, please provide the different estimates.

15) During the review of the WOG Topical and the associated pilot application, it was expected and observed that segments would be distributed throughout the four regions on the WCAP Structural Element Selection Matrix. Your evaluation, however, resulted in only four redundant segments being placed in Region 1.

a) Please describe the sequence and timing of events leading to core damage and large early release following the failure of the four segments in Region 1 (rupture of the charging seal injection lines).

b) Please provide the four, without ISI, estimated risk measures (the CDF with and without operator action, and the LERF with and without operator action) for these segments.

c) The submittal states that Westinghouse Owner's Group Peer Review Certification was conducted for the MP3 Probabilistic Risk Assessment (PRA) model in 1999. Please provide any Observation and Fact sheets regarding the Accident Sequence Evaluation subelements supporting the LOCA sequences analyses, and on the sequences used to model the rupture of the charging seal injection lines.

16) The submittal states that "at least one" structural element per HSS segment in the reactor coolant loop piping will be inspected. How many segments per loop are HSS and how many are Low Safety Significance (LSS)? How many volumetric inspections will done in each reactor coolant loop?

17) What criteria did you use to differentiate between High Failure Importance and Low Failure Importance in Figure 3.7-1 of WCAP-14572 Rev. 1-NP-A? Please include the break size and frequency (or 40 year probability).

18) The failure probability estimates used to support the statistical analysis are developed specifically to meet the statistical model input parameter definitions. These parameters are different from the failure estimate parameters used in support of the segment ranking and change in risk calculations, and it is expected that the value of the parameter would also be different. If your methodology deviated from the Topical report, please describe and justify your criteria and calculations:

Please confirm that the "Probability of a [unacceptable] Flaw" and "Conditional Probability of Leak/Year/Weld" are calculated for MP3 using the SRRA code as described on page 171 of the Topical report. How many SRR calculations were made to support the statistical analysis? Please confirm that the suggested probability of detection of 0.2 and the "Target Leak Rate/Year/Weld" as provided in Table 3.7-1 were used.