Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 5, 1992

Docket No. 50-423

Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

SUBJECT: STAFF EVALUATION OF MILLSTONE 3 INDIVIDUAL PLANT EXAMINATION (IPE) - INTERNAL EVENTS, GL 88-20 (TAC NO. M74434)

The purpose of this letter is to transmit our evaluation of the internal events portion of your Independent Plant Examination (IPE) that you submitted August 31, 1990, in response to Generic Letter 88-20.

Northeast Nuclear Energy Company (NNECO) responded to Generic Letter 88-20 and its supplements in letters dated April 22, 1991 and December 23, 1991.

The NRC staff completed its review of the internal events portion of the IPE submittal, and associated documentation which includes the Millstone 3 Station Probabilistic Safety Study (PSS), and NNECO responses to staff generated questions seeking clarification of their IPE process. No specific unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the Millstone 3 IPE.

The Millstone 3 IPE did not identify any severe accident vulnerabilities associated with either core damage or unusually poor containment performance. However, the PSS/IPE did identify improvements, all but one of which NNECO has already implemented. These improvements focus on reducing both core damage frequency and offsite release of radioactivity.

The staff notes, however, that the Millstone 3 IPE reported a smaller (by more than an order of magnitude) loss of offsite power contribution to core damage than that estimated in previous staff studies on station blackout. However, NNECO has committed to install a third air-cooled diesel generator in accordance with the Station Blackout requirements. Because implementation of this third diesel should reduce the loss of offsite power/blackout contribution, the staff did not pursue this difference further during its review.

Based on the Step 1 review of the Millstone 3 IPE submittal and previous staff reviews of the PSS which include reviews by both Brookhaven National Laboratory (BNL) and Lawrence Livermore National Laboratories (LLNL), the staff concludes that, with installation of the third diesel, NNECO met the intent of Generic Letter 88-20.

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A separate safety evaluation will be provided to document the review of the external event portion of the Millstone 3 IPE submittal. By this letter we are closing TAC Number M74434.

Sincerely,

/s/

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ENCLOSURE 1

STAFF EVALUATION OF MILLSTONE 3 INDIVIDUAL PLANT EXAMINATION
(IPE)

(INTERNAL EVENTS ONLY)

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EXECUTIVE SUMMARY

The NRC staff completed its review of the internal events portion of the IPE submittal, and associated documentation which includes the Millstone 3 Station Probabilistic Safety Study (PSS), and licensee responses to staff generated questions seeking clarification of their IPE process. No specific unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the Millstone 3 IPE.

The 1983 full-scope Level 3 Millstone 3 PSS (and subsequent updates) form the basis of the licensee's IPE. Six substantial PSS updates followed the completion of the study. The licensee PSS update process includes an initial screening of all plant changes, followed by a more detailed review and revision of plant models as appropriate. The licensee established updating procedures in early 1988 which now require probabilistic safety assessment (PSA) engineers to review and prioritize design changes to the plant. Updates also involve frequent exchange of information between the operation staff and PSA staff. The licensee plans to keep the PSS "living," via the employment of an integrated PC-based PSA model.

The Millstone 3 IPE did not identify any severe accident vulnerabilities associated with either core damage or "unusually poor" containment performance. However, the PSS/IPE did identify improvements, all but of which the licensee has already implemented. These improvements focus on reducing both core damage frequency and offsite release of radioactivity.

The staff notes, however, that the Millstone 3 IPE reported a smaller (by more that an order of magnitude) loss of offsite power contribution to core damage than that estimated in previous staff studies on station blackout. However, the licensee has committed to install a third air-cooled diesel generator in accordance with the Station Blackout requirements. Because implementation of this third diesel should reduce the loss of offsite power/blackout contribution, the staff did not pursue this difference further during its review.

Based on the Step 1 review of the Millstone 3 IPE submittal and previous staff reviews of the PSS which include reviews by both Brookhaven National Laboratory (BNL) and Lawrence Livermore National Laboratories (LLNL), the staff concludes that, with installation of the third diesel, the licensee met the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in Generic Letter 88-20; (2) the front-end systems analysis, the back-end containment performance analysis, and human reliability analysis are capable of identifying plant-specific vulnerabilities to severe accidents; (3) the licensee employed a viable means (review of applicable plant design change records, updating models as appropriate, and plant walkdowns) to verify that the IPE reflected the current plant design and operation; (4) the PSS which formed the basis of the IPE had an extensive independent peer review; (5) the licensee participated fully in the IPE process consistent with Generic Letter 88-20; (6) the licensee appropriately evaluated Millstone 3 decay heat removal function for vulnerabilities (resolving USI A-45); and (7) the licensee responded appropriately to the recommendations stemming from the Containment Performance Improvement (CPI) program. In addition, the licensee is actively using the IPE as a "living" document to enhance plant safety.

The staff's review is a process review which, in general, is not intended to validate the accuracy of the licensee's IPE findings. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on the licensee's ability to examine Millstone 3 for severe accident vulnerabilities, and not specifically on the detailed findings (or quantification estimates) which stemmed from the examination.

I. BACKGROUND

On November 23, 1988, the NRC issued Generic Letter 88-20 (Ref. 1) which requires licensees to conduct an Individual Plant Examination in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, licensees are expected to (1) develop an overall appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur at its plant, (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335 (Ref. 2), all licensee IPEs are to be reviewed by NRC teams to determine the extent to which licensees' IPE process met the intent of Generic Letter 88-20. The IPE review itself is a two step process; the first step, or "Step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "Step 2" review. The decision to go to a "Step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous PSA experiences. A unique design may also warrant a "Step 2" to better understand the implication of certain IPE findings and conclusions. As part of this process, the Millstone 3 IPE only required a "Step 1" review.

The "Step 1" review of the Millstone 3 IPE submittal involved an examination of the submittal, formulation of questions for additional information, meeting with the licensee to better understand the licensee's involvement, and consolidation of IPE insights and findings for data base storage. This review is limited in scope as it is designed to look for significant omissions, or inconsistencies with commonly accepted Probabilistic Safety Assessment (PSA) practices. The review process is not intended to validate the accuracy of the licensee's IPE, nor the numerical results generated as part of the analytic process.

The staff review of Millstone 3 from a analytic perspective began in 1983 when Northeast Utility staff and analysts from Westinghouse completed and submitted to the NRC staff a full-scope Level 3 risk assessment of the Millstone 3 plant entitled: "Millstone 3 Probabilistic Safety Study" (PSS) (Ref. 3). The study contained a full range of both internal and external event PSA models. The NRC subsequently contracted Lawrence Livermore

National Lab (LLNL) to review the core damage models (Ref. 4) and Brookhaven National Lab (BNL) to review the containment performance analysis (Ref. 5). Following the original PSS effort, six substantial PSS updates were performed by the licensee (see Attachment 1). Since that time, the licensee has maintained the PSS as a "living" document, and employed the PSS as the basis for their IPE.

On August 31, 1990, Northeast Utilities formally submitted the Millstone 3 IPE (Ref. 6) in response to Generic Letter 88-20 and associated supplements (Ref. 7-9). The IPE submittal contains the results of an evaluation of both internal and external events; however, the staff only reviewed the internal events portion. The external events portion will be reviewed separately, within the framework prescribed in Generic Letter 88-20 Supplement 4 (Ref. 10). The NRC review team subsequently met with the licensee on November 7, 1990, to discuss the Millstone 3 IPE findings and conclusions. Following the review of the IPE submittal and associated information, the IPE team generated and formally sent questions to the licensee seeking additional information and clarification (Ref. 11). The licensee responded to the staff's request in a letter dated April 22, 1991.

The following list summarizes the information reviewed during the evaluation of the licensee's IPE:

1. Millstone 3 response to Generic Letter 88-20 (Ref. 6)
2. Millstone 3 response (Ref. 12) to NRC request for additional information and subsequent telephone response to NRC's questions
3. Millstone 3 Station Probabilistic Safety Study (PSS) (Ref. 3)
4. Report by Lawrence Livermore National Laboratory (LLNL) (NUREG/CR-4142) (Ref. 4) on review of the Millstone 3 PSS
5. Millstone 3 risk evaluation report by Brookhaven National Laboratory (BNL) (NUREG/CR-4143) (Ref. 5)
6. Staff report on review of the Millstone 3 PSS (NUREG-1152) (Ref. 13)

The report documents findings and conclusions which stemmed from the NRC review. Specific numerical results and other insights taken from the licensee's IPE submittal are listed in the appendix.

II. <u>STAFF'S REVIEW</u>

1. <u>Licensee's IPE Process</u>

The Millstone 3 IPE submittal describes the approach taken by the licensee to confirm that the IPE represents the current as-built as-operated plant. The process includes review of applicable plant design change records, updating models as appropriate, and plant walkdowns. This process has been proceduralized as part of the risk management process at Millstone 3. The licensee intends to continue the process and maintain the IPE/PSS as a "living" document.

The staff examined the information associated with the licensee's walkdown activities. The IPE submittal notes that utility personnel and contractor personnel performed plant walkdowns of all modelled systems and plant areas including containment. During these walkdowns, the licensee performed a check of the modelling with the as-built and as operated plant information. The licensee indicated that plant walk-throughs and interaction with plant operations personnel are routine activities whenever situations at the plant require PSA staff input. Based on the review of the IPE and associated documentation, the staff finds that plant design change record reviews in conjunction with plant walkdowns, constituted a viable process capable of confirming that the IPE represents the as-built, as-operated plant.

The IPE submittal contains a summary description of the licensee's staff participation in the IPE process and the subsequent in-house peer review of the final product. The staff reviewed the licensee's description of the IPE program organization, composition of the peer review teams, and peer findings and conclusions. The staff notes that utility personnel participated fully in the IPE process, and that an extensive peer review had been performed of the original PSS which forms the basis of the IPE submittal.

Quantitative contributions to core damage frequency (CDF) by functional sequences, initiating events, individual systems, and individual operator actions, formed the licensee's basis for evaluating potential core damage vulnerabilities. The CDF functional sequences examined totaled 7E-5/reactor-year (RY), of which 80% resulted from internal events. The licensee did not formally define "vulnerability," but specified conditions that would generally be associated with a "major vulnerability," i.e., significant single failures, common cause failures or operator actions that have a high impact on core damage frequency; support systems with a relatively high probability of causing a plant transient and multiple front-line and support systems failures; and containment failure modes with relatively high probability of occurrence relative to other large dry PWR containments. The licensee noted that a "major vulnerability" would "necessitate action" up to and including plant shutdown. Less significant vulnerabilities would be addressed on a cost-effective basis.

The Millstone 3 IPE did not discover any major vulnerabilities, although "minor" vulnerabilities were identified and addressed as discussed in the licensee's response (Ref. 12) to the staff generated questions. The staff notes, however, that the Millstone 3 IPE reported a smaller (by more that an order of magnitude) loss of offsite power contribution to core damage than that estimated in previous staff studies on Station Blackout (Ref. 13 and Ref. 16) However, the licensee has committed to install a third air-cooled diesel generator in accordance with the Station Blackout requirements. Because implementation of this third diesel should reduce the loss of offsite power/blackout contribution, the staff did not pursue this difference further during its review.

Based on the review of the Millstone 3 IPE submittal and installation of the third air cooled diesel, the staff notes the reasonableness of the licensee's IPE conclusion regarding identification and treatment of "vulnerabilities." The IPE was found to be complete with respect to the information requested in Generic Letter 88-20, and the PSS which formed the basis of the IPE had an extensive independent peer review. The staff finds the Millstone 3 IPE process capable of identifying severe accident vulnerabilities, and that such capability is consistent with the objective of Generic Letter 88-20.

2. Front-End Analysis

The staff examined the front-end analysis for completeness and consistency with acceptable PSA practices. The IPE referenced insights from the Surry plant as detailed in NUREG-1150 (Ref. 14), a design similar to Millstone 3. The IPE employed the support state large event tree model for the front-end analysis, and linked this model with the back-end containment rasponse model via 31 plant damage states. Event trees were developed based on functional headings. The staff notes the licensee's development of a PC-based version of the integrated plant model which is expected to be used to re-quantify future plant improvements and data. The staff finds the employed analytic approach consistent with the methods identified in Generic Letter 88-20 for use in the IPE.

The initiating events appeared to have been appropriately reflected in the plant design dependency models and success criteria. The submittal contained 21 initiating events consistent with those generated in other PSAs and NUREG/CR-2300 (Ref. 15). The licensee initially ruled out (based on low probability) complete loss of service water as an initiator, but has subsequently agreed to include this event in a 1993 PSS update (Ref. 12). The licensee treated the service water system as a major support system in the IPE, and employed an event tree model with explicit illustration of the service water dependencies and mitigation actions. The staff finds treatment of the service water system consistent with the intent of Generic Letter 88-20, but agrees that follow-up studies would provide additional assurance that the IPE conclusions are correct with regard to failure of the service water system as a plant initiator.

The licensee explored plant specific initiators, (e.g., steam generator level control and biofouling of condenser cooling). The loss of instrument air was not modelled explicitly as a separate initiating event. The licensee did, however, consider and incorporate the impact of loss of air system as part of loss of Main Feedwater (MFW) event. The staff finds this reasonable (based on the Surry NUREG-1150 analysis) for closure of severe accident concerns, but also believes further insight, which might be useful in formulating emergency procedures or an accident management program, could be gleaned by treating loss of instrument air explicitly as a support system dependency in future PSS update activities.

The IPE submittal contained all front-line event trees, system and event tree success criteria, and support state event trees, and dependency matrix. The implications of two support systems were questioned further as part of the review process: (1) Heating Ventilation and Air Conditioning (HVAC) and (2) certain DC power dependencies.

Recent PSA studies have treated loss of HVAC explicitly within their framework. In response to a reviewer's question (Ref. 12), the licensee stated that loss of HVAC is not a "significant" core damage issue, based on the design, improved room cooling reliability in response to the Station Blackout Rule, and operators' awareness of potential equipment failure due to high temperature. The staff finds this rationale reasonable in meeting the intent of Generic Letter 88-20, however, the staff believes that explicit modelling of HVAC would provide additional insights and certainty in plant behavior during situations involving loss of HVAC. These insights might be useful in formulating emergency procedures or as input into the accident management program.

With regard to DC power, the licensee acknowledged that the PSS model lacked explicit illustration of DC power dependencies and dependencies on DC power in the support system model, but stated that dependencies were considered implicitly. The licensee provided a system dependency matrix to illustrate dependencies on DC power, and agreed to update (Ref. 12) the analysis with explicit treatment of DC power following completion of IPEs for other units (1993). In addition, the consideration of loss of a single DC bus, and loss of all vital DC power as special initiators formed the licensee's basis that no "risk outliers" are associated with the DC system. The staff finds the basis consistent with the intent of Generic Letter 88-20, and agrees that explicit illustration and documentation of DC dependencies in subsequent updates would be beneficial.

The Millstone 3 IPE considered four types of common cause failures (CCFs): (a) support system failures, (b) command failures, (c) human errors, (d) environmental conditions. CCFs resulting from the support system failures and the command failures were explicitly treated through the support system event tree modelling. The residual CCFs due to human errors and environmental conditions were addressed using the Binomial failure rates (BFRs) and have been explicitly modelled into the system fault trees. The staff notes the licensee's analytic treatment of CCF is consistent with NUREG/CR-2300 [PRA Procedures Guide] .

With only limited operating experience (about four reactor-years), the Millstone 3 IPE utilized generic data for most of the components in the system models. Generic sources include the Westinghouse Nuclear Technology Division Proprietary Data Base and the National Reliability Evaluation Program (NREP) Data Base. The licensee performed specific calculations for the diesel generator failures. The licensee also collected plant-specific data for loss of main feedwater (MFW) event, turbine trip, reactor trip, and the primary to secondary power mismatch events. For all ASME Class 1, 2 and 3 pumps and valves at Millstone 3, the demand failure probabilities were updated to account for the revised test intervals and test frequencies as documented in the Millstone 3 inservice test pump and valve program. For loss of AC power events, the licensee performed a Bayesian update employing data collected by Oak Ridge National Laboratory (ORNL) to quantify the IPE.

For a plant with limited operating experience, the staff finds the use of generic data reasonable, but also believes that the licensee would benefit from examination of future plant-specific information for potentially unrecognized component failure modes and sequences. In addition, the staff believes that validation of maintenance unavailabilities against plant-specific information would also help assure that employed generic unavailability estimates are being met.

The licensee did not develop event trees explicitly for the internal flood evaluation, but employed a screening evaluation using details developed as part of the Appendix R-related activities. The licensee considered a fire zone as a flood zone and performed a zone-specific flood cause-impact analysis by quantifying the flood initiating event frequency based on pipe locations, flood sources, location of safety system components, and inter-zone flood propagation information. The licensee used a screening analysis of the zone-specific floods to determine whether a postulated flood could cause an initiating event and/or could affect one or more trains of a mitigating safety system. Because of the physical separation of systems and the small impact of flooding on multiple (diverse) means of decay heat removal, the IPE found a low probability of internal flood induced core damage (8.5E-7/yr) at the screening level.

The staff finds the treatment of internal flood reasonable for addressing potential "outliers", but also believes that the licensee could benefit from investigating further the potential for inter-zone flooding, e.g., check valve failures inside drain systems, and maintenance activities which could compromise flood barriers. Further study would be useful in formulating emergency procedures or as input into the accident management program by providing insight into flood initiators and potential recovery actions.

The submittal contains the top 100 most probable core damage sequences accounting for 96% of the total mean CDF due to internal events (totalling 5.5*E-5/RY). The sequences identify loss of coolant accidents (large and medium LOCAs) as dominating the core damage frequency (34%), with steam line breaks (14.7%) and loss of offsite power (9%) as other dominating sequences. The IPE identified unavailability of the recirculation function due to common cause motor operated valve failures as the dominant contributor to the LOCA sequences, whereas failure to depressurize the primary system (due to high operator error and random failure of relief valves) contributed to the steam line break sequence.

The staff notes that previous staff studies (Ref. 13 and Ref. 16) found Millstone 3 to have a much higher (i.e., over an order of magnitude) contribution to core damage from station blackout than previously reported in the PSS or the IPE submittal. As part of complying with the Station Blackout Rule (10CFR50.63), the licensee committed to installing a third air-cooled diesel generator to reduce the loss of offsite power/blackout contribution. (It should be noted that the licensee's IPE did not take credit for the third diesel.) Because implementation of this third diesel is expected to reduce the loss of offsite power/blackout contribution, the staff did not pursue this difference further during its review.

Millstone 3 dominant sequences are conditioned on the implicit fact that internal events involving reactor cooling pump (RCP) seal failure are found in sequences totaling only 10% of the overall core damage frequency estimate. The licensee employed analytical and experimental information from the Westinghouse Owners Group in investigating RCP seal failure, and implemented procedures that would establish once through cooling of charging pump during loss of service water. Although the staff did not examine the RCP model in detail (the licensee did not attempt to resolve the associated Generic Issue [23] in the IPE), licensee action in response to insights from their evaluation (RCP failure in conjunction with loss of charging pump cooling), is consistent with the intent of the Generic Letter 88-20.

With recognition of the installation of the air-cooled diesel, the staff found the licensee's front-end IPE analysis complete, with the level of detail consistent with the information requested in NUREG-1335 (Ref. 2). In addition, the employed analytical techniques are consistent with other NRC reviewed and accepted PSAs and capable of identifying potential core damage vulnerabilities. The staff, therefore, finds the IPE front-end analysis met the intent of Generic Letter 88-20.

3. <u>Back-End Analysis</u>

The staff examined the back-end analysis for completeness and consistency with other acceptable PSA practices. Millstone 3 utilizes a large (2.3*E+6 ft^3) subatmospheric containment structure, about 28% larger than Surry. Plant specific structural analysis determined a median failure pressure of 117 psig with the 5th and 95th percentile values for containment failure pressure of 97 and 132 psig respectively.

The staff examined the licensee's documentation of referenced codes, analytical models and input data. The back-end analysis utilized methodology similar to that exercised in the Zion and Indian Point PSSs. The MARCH computer code modelled in-vessel severe accident phenomenon, the MODMESH computer code modelled reactor pressure vessel blow-down, CORCON-MOD1 modelled molten core-concrete interaction and the Westinghouse COCOCLASS 9 modelled containment thermal response.

The Level 1 analysis resulted in the identification of 31 plant damage states which were subsequently binned into 10 core damage containment response classes, plus an additional group representing containment bypass plant damage states for which containment response was not required. Containment Event Trees (CETs) were developed for the plant damage states and divided into 6 distinct accident progression time frames consisting of 17 nodes. The CET end points were subsequently binned into 13 distinct release categories. CORRAL-II code determined radionuclide release fractions for the 13 release categories.

The licensee defined "unusually poor" containment performance (UPCP) as those events resulting in early containment failures, containment bypass failures or containment isolation failures. By this definition the frequency of UPCP was reported to be 4.17*E-7/RY with a conditional probability of UPCP of 7.55*E-3. The IPE/PSS estimated containment isolation failure probability to be 2.0*E-4, resulting in a commensurate frequency of core melt with containment isolation failure of 1.1*E-8/RY. These low values for containment isolation failure are attributed primarily to the inherent characteristics of subatmospheric containment operation.

The licensee considered failure of elastomer material primarily used to seal personnel and equipment hatches and electrical penetration assembles. The mechanical and thermal properties of the elastomer seals enabled seal failure pressures to be in excess of the failure pressures predicted by the structural analyses. Heat transfer, mass transport analyses, and evaluation of maximum leakage areas afforded by clearances between metal to metal contacting surfaces were utilized to support the above conclusion.

Because of similarities in design, the licensee made extensive comparisons of the Millstone 3 PSS results and insights with the results and insights from the NUREG-1150 analysis of the Surry 1 facility. The two plants have many similarities with regard to containment characteristics. In particular the subatmospheric containment design, reactor cavity configuration and concrete types are generally the same. The most pronounced difference is in the refueling water storage tank (RWST) capacity where Millstone 3 has several times the volume of Surry. In general, most of the insights from the Surry 1 Probabilistic Risk Assessment (PRA) are consistent with the corresponding insights from the Millstone 3 PSS. The two most significant exceptions concern the instrument room/seal table room layout and the potential for reflood of the reactor cavity from the containment sump.

For both plants the seal table is located in an area/room inside the crane wall. Unlike Surry, however, the Millstone 3 design passageway through the crane wall allows a 22 foot long direct line of sight from the seal table area to the containment wall. In the event of a high pressure melt ejection (HPME) molten core debris could reach the instrument/seal table rooms via the reactor cavity and instrument tunnels. The Millstone 3 design would, therefore, present an additional potential for containment failure due to molten core debris attack of the containment wall. However, for this scenario the probability of containment failure due to direct containment heating (DCH) is already high, and dominates the failure mode.

With regard to reflood of the reactor cavity, differences include the significantly greater RWST volume of Millstone 3 over Surry (2.3*E+6 gal. vs 3.5*E+5 gal.), increasing the likelihood of cavity flooding for Millstone 3. This results in some differences in the characterization and timing of containment response to accident phenomena, but both plants exhibit a rather high conditional probability of no containment failure (about .8).

The Millstone 3 PSS, in agreement with the Industry Degraded Core Rulemaking (IDCOR) evaluation, is based upon the expert opinion that the reactor cavity/instrumentation tunnel configuration is expected to retain essentially all of the core debris during a high pressure melt ejection severe accident. The cavity area geometry is also expected to reduce the potential for establishing effective air currents between the cavity and general containment volume for heat removal from core debris in the cavity area. This is consistent with NUREG-1150 Surry 1 PRA (Ref. 14). However, since the development of both the Surry 1 and Millstone 3 PRAs, small (1/42nd) scale experiments at BNL appear to contradict this conclusion for the Surry plant. For this reason the licensee is participating in industry sponsored research to address the HPME/DCH issue and has identified necessary experiments to provide specific insights for Millstone 3. The licensee recognizes the potential for the DCH issues to significantly increase the probability of early containment failures. The licensee's "living" PSA program, however, provides a means by which to incorporate insights from on-going severe accident research on the HPME/DCH phenomena into the Millstone 3

PSS as needed. The licensee has concluded that complete reanalysis of the back-end would be premature at this time. The staff finds this approach reasonable, considering the large uncertainties intrinsic to back-end analyses, and the belief that complete reanalysis is not expected to change the IPE conclusions regarding containment vulnerabilities.

The licensee did not find any vulnerabilities that would lead to unusually poor containment performance (UPCP). However the licensee did note the sensitivity of early containment failure (one component of UPCP) to the uncertainty of the reactor cavity/instrument tunnel debris retention characteristics. As discussed above, the licensee's intent to keep the PSA program "living" will allow the licensee to modify the back-end analysis to accommodate improved perception of transport characteristics (which may result from severe accident research activities), and obtain further insight into the significance of HPME/DCH at such time when the results of research are sufficiently clear to warrant reanalysis.

In summary, the 1983 Millstone 3 PSS which forms the basis of the licensee's IPE, has been amended and augmented to incorporate revised methodology, current plant configuration, and current equipment performance characteristics. Specifically the licensee's IPE addressed the most important severe accident phenomena normally associated with large dry containments ie, DCH, Induced Steam Generator Tube Rupture (ISGTR) and hydrogen combustion. The IPE review did not identify any obvious or significant problems or errors in the back-end analysis. The overall assessment of the back-end analysis is that the licensee has made reasonable use of PSA techniques in performing the back-end analysis, and that the techniques employed are capable of identifying severe accident vulnerabilities. Based on these findings the staff concludes that the licensee's back-end IPE process is consistent with the intent of Generic Letter 88-20.

4. Human Factor Considerations

The 1983 Millstone 3 PSS and associated human reliability analysis (HRA) formed the basis for the treatment of human error in the Millstone 3 IPE. Lawrence Livermore National Laboratory (LLNL) reviewed the HRA and published their findings in April, 1986 (Ref. 4). In their report, LLNL concluded that the HRA was performed "in a reasonable and consistent manner in keeping with the methods suggested in the NREP Procedures Guide and NUREG/CR-2815." The review, however, identified and analyzed three human errors not included in the Millstone HRA: (1) operator overthrottles high pressure injection (HPI) resulting in inadequate flow, (2) operator erroneously terminates HPI, and (3) operator fails to control HPI during SGTR. In response, the licensee updated their PSS/HRA (in 1987 as part of Amendment 4) accordingly, and presently continues to maintain the HRA as a "living" document.

The Millstone 3 HRA analyzed human actions based on the method prescribed in NUREG/CR-2815, Rev. 1 (Ref. 17), an approach the NRC found acceptable for use in the IDCOR Individual Plant Examination Methodology. The employed technique modelled human recovery actions and human errors on the event trees and the fault trees respectively, and considered whether human actions and associated errors are cognitive-based or procedural-based. Human actions were quantified using time reliability curves contained in NUREG/CR-2815. In addition, the licensee exercised different HRA methods as part of their overall IPE/PSS "living" program to update and address human reliability. Working within the "Systematic Human Action Reliability Procedure" (SHARP) framework, the licensee employed the behavioral model from Appendix A of EPRI NP-3583, the Human Cognitive Reliability (HCR) model, and post event human decision errors: operator action tree/time reliability correlation. Techniques for Human Error Rate Prediction (THERP-Ref. 18) analyses were performed for the pre-accident human actions identified in the screening analysis. The staff finds these methods consistent with the intent of Generic Letter 88-20, in that they allowed the licensee to develop a quantitative understanding of the contribution of human error to core damage and identify dominant sequences.

Pre-accident human errors identified in the Millstone study generally stem from failure to restore equipment to the proper position after test or maintenance. Important post-accident human actions identified by the licensee in order of significance include:

a. transfer to sump recirculation,
b. primary feed and bleed,
c. recovery of main feedwater,
d. emergency boration,
e. recovery of off-site power,
f. controlled primary depressurization,
g. secondary depressurization to low pressure safety injection (LPSI) shutoff pressure.

These findings are consistent with other PSAs.

Insights from the original HRA and other follow-on studies have been incorporated into plant procedures at Millstone 3. The IPE submittal stated that a number of procedural changes have been implemented based upon probabilistic insights, the most significant involving station blackout and interfacing systems LOCA sequences. The licensee also increased emphasis on operator training, addressing scenarios involving containment sump recirculation, LOCA outside containment, and feed and bleed. These modifications illustrate that the licensee appropriately considered human actions in their efforts to reduce the CDF and improve plant safety.

Based on the information contained in the earlier PSS/HRA (and updates), staff review reports, the IPE submittal, and responses to staff questions as part of the IPE review effort, the staff finds the HRA process employed at Millstone 3 capable of discovering severe accident vulnerabilities from human errors consistent with the intent of Generic Letter 88-20. In addition, the licensee's intent to maintain a "living" IPE program will provide additional assurance that the licensee will continue to evaluate potentially important human actions as they are identified.

5. Containment Performance Improvements (CPI)

Generic letter 88-20 Supplement 3 (Ref. 9) contains CPI recommendations which focus on the vulnerability of containments to severe accident challenges. For large dry containments, such as the Millstone 3 design, the CPI program results recommend that licensees in their IPE consider hydrogen production and control during severe accidents, particularly on the potential for local hydrogen detonation.

Containment failures, due to containment overpressure resulting from global hydrogen combustion, have been incorporated into the Millstone 3 PSS for sequences in which the containment volume is deinerted as a result of continuous containment spray operation, or recovery of containment spray following loss of AC power. The potential for global detonation of hydrogen is considered to be negligibly small. This is consistent with the NUREG-1150 findings for Surry and Zion and the conclusions reached in the IDCOR program. Also, as a result of a review and analysis of the Millstone 3 containment design, site walkdowns, and comparisons to the Surry and Zion containment designs, the licensee concluded that there is a negligible probability of containment failure or severe damage that could result from local detonations due to hydrogen "pocketing" inside containment. The licensee based this conclusion upon the open containment features, minimal enclosed spaces and the liberal use of open floor gratings. Furthermore, as a result of previous BNL and NRC staff reviews of the Millstone 3 PSS, the licensee evaluated potential containment performance improvements, including pros and cons of containment spray recovery, manually operated AC independent containment spray system, hydrogen igniter system and flooded reactor cavity configuration. The evaluation did not identify any containment improvements which were of sufficient significance or cost effectiveness to warrant implementation.

The licensee's conclusions are consistent with those from NUREG-1150 Surry 1, which also has a subatmospheric containment design that closely resembles that of Millstone 3. The staff, therefore, concludes that the licensee's response to CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of Generic Letter 88-20 and associated Supplement 3.

6. Decay Heat Removal (DHR) Evaluation

In accordance with the resolution of USI A-45 "Shutdown Decay Heat Removal Requirements," the licensee performed an examination of the Millstone 3 DHR system to identify vulnerabilities. The licensee's examination included a DHR function evaluation during LOCA events (with emphasis on small LOCA events) and transients. Examination of plant-specific DHR features included the MFW system, auxiliary feedwater (AFW) system, feed and bleed operation, recirculation system, and RWST capability. As part of future efforts, the staff also believes that emergency operating procedures and the licensee's accident management could potentially benefit from investigating further the impact of loss of Turbine Building Component Cooling Water on the DHR function, although it is not expected to change the IPE conclusions regarding DHR reliability.

The licensee utilized insights gained from NRC sponsored PSAs (summarized in Appendix 5 to Generic Letter 88-20). Redundancy and diversity in the front-line and support systems significantly reduced the DHR function as a contributor to core damage at Millstone 3. This was noted in the licensee's response to a reviewer's questions, in which the licensee performed a sensitivity study that indicated without feed and bleed cooling and recovery of main feedwater, core damage frequency would increase more than an order of magnitude (to approximately 7E-04/yr.).

The following were noted by the licensee to have increased DHR reliability at Millstone 3:

(a) three redundant trains of AFW (two motor driven, one steam driven)
(b) three MFW pumps (one motor driven, two steam driven)
(c) assignment of high priority to feed-and-bleed operator training
(d) diversity and redundancy in systems utilized for recirculation
(e) larger capacity RWST to extend time available before going into recirculation
(f) special main control room features to reduce errors of commission associated with recirculation.

Based on the licensee's process used to search for DHR vulnerabilities, and review of plant-specific features, the staff concludes that the licensee's DHR evaluation is consistent with the intent of Generic Letter 88-20 to resolve USI A-45. Therefore the staff finds USI A-45 resolved for Millstone 3.

7. Licensee Actions and Commitments from the IPE

In addition to the licensee intent to maintain a "living" IPE/PSS program, the submittal documents the licensee's use and planned future use of the IPE/PSS (some of which extend beyond the original intent of the program):

a) operator training in risk dominant sequences
b) safety evaluations
c) establishment of equipment test intervals
d) prioritization of important equipment and systems
e) establishment of allowable outage times for safety related equipment.

The IPE submittal contains a discussion of improvements which have been analyzed for their cost-benefit as a result of the PSS and IPE. Although some of the improvements had been initiated for regulatory reasons, the licensee states that the PSS often provides further motivation for implementation.
Plant improvements noted by the licensee to have "measurable improvement in core melt frequency and/or public safety" and already implemented include:

1. Installation of an anticipated transient without scram (ATWS) mitigation system to provide alternate means of turbine trip and actuation of AFW.
2. Modifications to the main control board to reflect "transfer to cold leg recirculation emergency operating procedure (EOP)."
3. Implementation of once-through-cooling of charging pumps for loss of service water events.
4. Implementation of a part stroke test of accumulator check valves every refueling interval.
5. Procedure modifications to ensure sufficient water in containment recirculation pumps.
6. RHR autoclosure interlock removal to eliminate a major contributor to RHR system unavailability during shutdown and alarm installation to reduce potential for interfacing system LOCA.

In addition, the licensee plans to add a third diesel generator for safe shutdown loads during loss of off-site power (To be implemented in accordance with Station Blackout Requirements).

Although the team did not examine the merits of the above improvements in detail, the staff notes that the licensee is applying PSS/IPE findings to enhance plant safety consistent with the intent of Generic Letter 88-20. The staff, therefore, finds the licensee's actions and commitments reasonable for closure of severe accident concerns.

III. CONCLUSION

The staff concludes that the licensee's internal events portion of the IPE process is consistent with the intent of Generic Letter 88-20. The staff based this conclusion on the following findings:

1. The licensee employed a viable process capable of confirming that the IPE represents the as-built, as-operated plant. The process included performing plant design change record reviews in conjunction with plant walkdowns and PSS updates.

2. With recognition of the installation of the air-cooled diesel, the front-end IPE analysis appears complete, with the level of detail consistent with the information requested in NUREG-1335. In addition, the employed analytical techniques are consistent with other NRC reviewed and accepted PSAs and capable of identifying potential core damage vulnerabilities.

3. The licensee's IPE addressed the most important severe accident phenomena normally associated with large dry containments ie, DCH, Induced Steam Generator Tube Rupture (ISGTR) and hydrogen combustion. No obvious or significant problems or errors in the back-end analysis were identified. The licensee has made reasonable use of PSA techniques in performing the back-end analysis, and that the techniques employed are capable of identifying severe accident vulnerabilities associated with containment failure.

4. Using techniques consistent with other NRC reviewed and accepted PSAs, the Human Reliability Analysis (HRA) allowed the licensees to develop a quantitative understanding of the contribution of human error to core damage and was capable of discovering severe accident vulnerabilities from human errors.

5. Based on the licensee's process used to search for DHR vulnerabilities, and review of Millstone 3 plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of Generic Letter 88-20 to resolve USI A-45.

6. The licensee's conclusions are consistent with those from NUREG-1150 Surry 1, which also has a subatmospheric containment design that closely resembles that of Millstone 3. The licensee's response to CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of Generic Letter 88-20 Supplement 3.

In addition, licensee personnel participated fully in the IPE process, and an extensive peer review had been performed of the original PSS which forms the basis of the IPE submittal. The licensee also plans to use the IPE as a "living" document which will enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the lifetime of the plant.

The staff notes, however, that the Millstone 3 IPE reported a smaller (by more that an order of magnitude) loss of offsite power contribution to core damage than that estimated in previous staff studies (NUREG-1152, NUREG-1032). However, the licensee has committed to install a third air-cooled diesel generator in accordance with the Station Blackout requirements. Because implementation of this third diesel should reduce the loss of offsite power/blackout contribution, the staff did not pursue this difference further during its review.

Based on the overall review findings, the staff concludes that the licensee demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the Millstone 3 facility, has gained a quantitative understanding of core damage and fission product release, and with the commitment to install a third diesel generator, responded appropriately to safety improvement opportunities. The staff, therefore, finds the Millstone 3 IPE process met the intent of Generic Letter 88-20. and is acceptable.

IV. REFERENCES

1. NRC letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20," dated November 23, 1988.

2. USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, Final Report, August 1989.

3. Northeast Utilities, "Millstone Unit 3 Probabilistic Safety Study," August 1983.

4. NUREG/CR-4142, "A Review of the Millstone 3 Probabilistic Safety Study," Lawrence Livermore National Laboratory, April 1986.

5. NUREG/CR-4143, "Review and Evaluation of the Millstone Unit 3 Probabilistic Safety Study," Brookhaven National Laboratory, September, 1985.

6. Letter from E. J. Mroczka of Northeast Nuclear Energy Company (NNECO), "Millstone Nuclear Power station, Unit No. 3 response to Generic Letter 88-20 Individual Plant Examination for Severe Accident Vulnerabilities Summary Report Submittal," August 31, 1990.

7. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20, Supplement No. 1," dated August 29, 1989.

8. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Accident Management Strategies for Consideration in the Individual Plant Examination Process - Generic Letter No. 88-20, Supplement No. 2," dated April 4, 1990.

9. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities except Licensees for Boiling Water Reactors with MARK I Containments, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20, Supplement No. 3," dated July 6, 1990.

10. NRC letter to All Licensees holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Generic Letter No. 88-20, Supplement 4," dated June 28, 1991.

11. USNRC Letter to Mroczka of NNESCO, "Millstone Unit 3 - request for Additional Information regarding Individual Plant Examination (TAC No. 74434)," January 17, 1991.

12. Letter from C. F. Sears of NNECO, "Millstone Nuclear Power station, Unit No 3, Response to Request for Additional Information regarding Individual Plant Examination (TAC No. 74434)," April 22, 1991.

13. NUREG-1152, "Millstone 3 Risk Evaluation Report," June 1986.

14. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Plants," (Second Draft for Peer Review), June 1989.

15. NUREG/CR-2300, "PRA Procedures Guide," January 1983.

16. NUREG-1032, "Evaluation of Station Blackout at Nuclear Power Plants," June 1988.

17. NUREG/CR-2815, Revision 1, "National Reliability Evaluation Program Procedures Guide," August 1985.

18. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," August 1983.

Attachment 1: A Chronology of Millstone 3 PSS Activities

Date	Description of the activity
Aug. 1983	Millstone 3 PSS submitted
Sept. 1983	Amendment 1: Corrected consequence analysis
Jan. 1984	Transfer of the PSS technology from Westinghouse, the PSS contractor, to the licensee
Apr. 1984	Amendment 2: Reanalysis of seismic fragilities by Structural Mechanics Associates
Nov. 1984	Amendment 3: Correction of mathematical error in seismic analysis
Aug. 1985	Published Millstone 3 risk evaluation report (NUREG-1152)
Aug. 1987	Amendment 4 (internal): Reanalysis of the Level 1 PRA to account for actual surveillance intervals, main feedwater recovery, etc.
1988	First round of evaluation of projects under internal integrated safety assessment program (ISAP)
1989	Second round of internal ISAP evaluations
1989-1990	Transferred PSS from mini-computer to PS/2
May 1990	PC version of PSS (5th update): Correction of math and logic errors discovered in transfer
June 1990	PC version of PSS (6th update): Updated transient frequencies (plant data), revised the V sequence, and coupled the Level 2 PRA to the Level 1.
Fall 1990	Coupled the Level 3 PRA to Levels 1 and 2; third round of ISAP evaluations
Aug. 1990	Submittal of the Millstone 3 IPE

APPENDIX
MILLSTONE 3 DATA SUMMARY SHEET*
(INTERNAL AND EXTERNAL EVENTS)

o Total Core Damage Frequency:

7.0E-5/year (mean value)
80% resulting from internal events
20% resulting from external events

o Major Initiating Events and contribution to core melt frequency (internal and external events):

	Contribution
Transients	
- LOOP	(9%)[1]
- Loss of 1 DC bus	(7%)
- Loss of 1 SW train	(5%)
Steam line break outside containment	(15%)
LOCAs	
- Large LOCA	(15%)
- Medium LOCA	(1[illegible]%)
- Small LOCA	(4%)
ATWS	(6%)
SGTR	(2%)

o Major contributions to dominant core melt sequences:

Medium and large LOCA sequences involving recirculation failures or common-cause failures of motor-operated valves in the service water system to recirculation heat exchangers.

Steam line break events (outside the containment) followed by failure to depressurize the primary system due to operator failing to open relief valves or random failures of these relief valves.

o Major operator action failures:

Transfer to sump recirculation
Primary feed and bleed
Recovery of main feedwater
Emergency boration
Recovery of off-site power
Controlled primary depressurization
Secondary depressurization to LPSI shutoff pressure

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[1] The concern regarding the difference between staff and licensee estimates of LOOP contribution to core melt frequency will be resolved by the installation of the additional emergency diesel generator.

o Conditional containment failure probability given core damage

Intact containment	80%
Late containment failure without sprays	11%
Basemat failure with sprays	4%
Late containment failure with sprays	3%
Basemat failure without sprays	<1%

o Significant PSA findings:

1. Station blackout is a major contributor to public risk.
2. Interfacing system LOCA is a major contributor to public risk.
3. Auxiliary feedwater and feed and bleed failures are in many accident sequences.
4. Failure of containment sump recirculation is found in dominant sequences.
5. Failure of safety injection accumulators is a major contributor to core melt frequency.
6. Loss of 125 V. vital DC power is a major contributor to core melt frequency.
7. Loss of 120 V. vital AC power is a major contributor to core melt frequency.
8. Dry lower reactor cavity results in substantial hydrogen generation.

o Plant modifications status based on PSA considerations:

1. Replacement of diesel generator lube oil cooler anchor bolts (installed).
2. Addition of ATWS mitigation system to provide alternate means of turbine trip and actuation of AFW (installed).
3. Changes to main control board to reflect "Transfer to Cold Leg Recirculation EOP," (installed).
4. RHR autoclosure interlock removal to eliminate a major contributor to RHR system unavailability during shutdown and alarm installation to reduce potential for interfacing system LOCA. (recently completed).
5. Improvement in monitoring RCS level, temperature, and RHR system performance during reduced inventory conditions (installed).
6. Provision of once-through cooling of charging pumps for loss of service water (implemented).
7. Provision of part stroke testing of accumulator check valves every refueling interval (implemented).
8. Modification of procedure to ensure sufficient water in containment recirculation pumps (implemented).
9. Addition of third diesel generator for safe shutdown loads (to be implemented in accordance with SBO requirements).

o Future Activities: Periodic update of PSS (the goal is to update the living PSA model on PC for every refuel cycle)

(* All information is taken from the Millstone 3 IPE and has not been validated by the NRC staff.)