UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSAL TO USE ASME CODE CASE N-560 AS AN ALTERNATIVE

TO ASME CODE, SECTION XI, TABLE IWB-2500-1

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NUMBER 50-271

1.0 INTRODUCTION

The Technical Specifications (TS) for Vermont Yankee (VY) Nuclear Power Station state that inservice inspection (ISI) of American Society of Mechanical Engineers (ASME) Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

By letters dated August 6, 1997, August 15, 1997, October 23, 1997, July 31, 1998, and September 4, 1998, Vermont Yankee Nuclear Power Corporation (VY or the licensee) pursuant to the provisions of 10 CFR 50.55a(a)(3), requested that the NRC staff approve the ASME Code Case N-560, "Alternative Examination Requirements for Class 1, Category B-J Piping Welds." The Code Case was proposed as an alternative to the requirements of ASME Code, Section XI Table IWB-2500-1 and relates to examination of Class 1, Category B-J, piping welds at the VY nuclear power station. Specifically, VY proposes to reduce the examination of ASME Code, Section XI, Category B-J welds from 113 welds to 41 welds and requested to use Code Case N-560 which permits this reduction as long as the licensee uses a probabilistic risk-informed approach in the selection of the welds. VY also has augmented the implementation of Code Case N-560 at the VY nuclear power station by using methodology approved by the Electric Power Research Institute (EPRI). The staff reviewed the proposed request as a site-specific request being applicable only to the VY nuclear power station.

In September 1998, the NRC issued Regulatory Guide (RG) 1.178 (draft was previously issued as DG-1063), "An Approach for Plant-Specific, Risk Informed Decisionmaking: Inservice Inspection of Piping." RG 1.178 augments the guidance presented in RG 1.174 (draft previously issued as DG-1061), "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis," by providing guidance specific to incorporating risk insights to inservice inspection programs for piping. The licensee's

request was reviewed on the merits of the submittals and general conformance with the regulatory guides.

2.0 SUMMARY OF LICENSEE'S PROPOSED APPROACH

The licensee has submitted a request to use ASME Code Case N-560 as an alternative to the requirements of ASME Code, Section XI Table IWB-2500-1. The Code of Record for the third interval at VY is the ASME Code, Section XI, 1986 Edition. The information provided by the licensee in support of the request has been evaluated and the bases for disposition are documented below.

2.1 Code Requirement

Section XI, Table IWB-2500-1, Examination Category B-J, Pressure Retaining Welds in Piping lists the examination requirements for Category B-J welds. Note 1(d) to the table states: "Examinations shall include the following: Additional piping welds so that the total number of circumferential butt welds (or branch connections or socket welds) selected for examination equals 25% of the circumferential butt welds (or branch connections or socket welds) in the reactor coolant piping system. This total does not include welds excluded by IWB-1220. These additional welds may be located in one loop (one loop is defined for both PWR and BWR plants in the 1997 Edition)."

2.2 Licensee's Code Relief Request

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee has requested an approval to use ASME Code Case N-560 as an alternative to the requirements of ASME Code Section XI Table IWB-2500-1. Code Case N-560 states inter alia: "The inspection program shall be based on a total number of examination zones consisting of not less than 10% of Class 1 (Category B-J) piping welds in each system, excluding socket welds, to be examined during each inspection interval. The selection process shall consist of the following:

> (1) Examination zones shall be selected based on a relative ranking process that identifies more risk-important segments in the system with regard to probability and consequences of failure. Examination zones shall be selected from those pipe segments that fall into the highest risk group.
>
> (2) The ranking process shall address relevant degradation mechanisms (e.g. corrosion, stress corrosion, thermal fatigue, thermal stratification, flow-accelerated corrosion) and industry failure experience with systems and components.
>
> (3) The consequences of failure at various locations in the system shall be based on the break size and operating mode that results in the highest impact on plant safety. Both direct and indirect effects shall be considered."

2.3 Licensee's Basis for Requesting Relief

The licensee, as stated in their letter dated October 23, 1997, provided the following basis to describe how the alternative program meets the intent of draft Regulatory Guide DG-1061 and Standard Review Plan 3.9.8:

"Draft Guide 1061, as well as Draft SRP Chapter 3.9.8 (Standard Review Plan For The Review Of Risk-Informed In-service Inspection of Piping) and Draft Regulatory Guide DG-1063 (An Approach For Plant-Specific, Risk-Informed Decision Making: Inspection of Piping), identified five principles of risk informed regulations. They are:

1. Meet current regulations,
2. Maintain defense in depth,
3. Maintain sufficient safety margins,
4. Proposed increase in risk (including cumulative effects) is small, NRC safety goals are not exceeded, and
5. Performance-based implementation and monitoring strategies.

Principle #1: Meet Current Regulations

10 CFR 50.55a and Appendix A to 10 CFR Part 50 are the primary regulations governing inservice inspection of piping. The intent of these regulations, as it pertains to the N560 scope of piping, is to assure a robust reactor coolant pressure boundary. Via reference in 10 CFR 50.55a, Section XI to the ASME B&PV Code, is the implementing vehicle for these inspections. Code Case N560 is an ASME approved alternative to current Section XI requirements. Other Section XI inspection activities such as the examination of Class 1 socket welded connections and dissimilar metal welds, pressure and leak testing requirements, Class 2 and 3 piping examinations, are not adversely affected by implementation of N560.

In addition to the other Section XI activities not adversely impacted by the implementation of N560 are the augmented inspection programs, such as in response to Generic Letter 88-01 (NRC Position on IGSCC in BWR Austenitic SS piping) and Generic Letter 89-08 (Erosion/Corrosion Induced Wall Thinning).

Other plant programs, which are not strictly inspection driven but can have a dominant impact on assuring piping reliability, include the primary water chemistry control program, reactor coolant leakage, drywell and feedwater nozzle monitoring efforts, all of which are unaffected by implementation of Code Case N560.

Principle #2: Maintain Defense in Depth

The intent of the inspections mandated by ASME Section XI for category B-J piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in the reactor coolant pressure boundary. Currently, the process for picking inspection locations is

based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 9201-01 Rev. 1 (Evaluation of In-service Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds in Piping), this method has been ineffective in identifying leaks or failures. In response to these findings ASME issued Code Case N560, which has a much more robust selection process founded on actual service experience with nuclear plant piping failure data.

The N560 selection process has two key ingredients. Those are 1) a determination of each location's susceptibility to degradation and 2) an assessment of the consequence of the location's failure. These two ingredients not only assure defense in depth is maintained, but actually increased over the current process. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leaks or ruptures in the reactor coolant pressure boundary is increased. Secondly, the consequence assessment effort has a single failure criterion so that, no matter how unlikely a failure scenario is, it is ranked high if, as a result of the failure, there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability so that less reliable equipment is not credited as much as more reliable equipment.

Principle #3: Maintain Sufficient Safety Margins

The safety function of interest in this evaluation is that of reactor coolant pressure boundary integrity. Listed below are those attributes necessary for fulfilling this requirement, as well as the impact of N560 on meeting the objective:

1. Quality Design - No Change
2. Quality Fabrication - No Change
3. Quality Construction - No Change
4. Quality Testing - No Change
5. Quality Inspection - Fewer inspections conducted at more appropriate locations using better techniques and, as necessary, expanded volumes. In addition, augmented inspection programs, such as IGSCC and FAC will continue.

As can be seen from the above summary, those attributes that are critical in defining and maintaining sufficient safety margins are unchanged except for a subset of the pressure boundary volumetric examinations. In this case, the augmented programs will continue, and the reduced number of volumetric Section XI exams are based upon the exceptional performance history of category B-J components. In addition, the new Section XI locations are more appropriate, usually involve larger inspection volumes, and will have better inspections conducted.

Principle #4: Proposed increase in risk (including cumulative effects) is small, NRC safety goals are not exceeded

This issue was addressed in two ways. The first way was a direct comparison of the old Section XI Program to the N560 recommended inspection. This comparison accounted for the

Conditional Core Damage Probability (CCDP) of each segment, its failure potential, and the impact of the inspection reliability. For this case, where the N560 inspection provides for a better inspection, credit is given in the form of a higher probability of detection.

The second way accounted for the CCDP of each segment and its failure potential, but did not credit the positive aspects of the N560 selection process and associated inspection for cause techniques. This assessment assumes that every segment within a risk category (e.g. Risk Category 2) had the highest identified CCDP for that category. This conservative estimate is then compared to the DG-1061 criteria for acceptability.

The results of this assessment are that the N560 application provides a net positive safety impact, even with a reduction in sampling locations from 25% to 10%. This is primarily due to three factors: 1) most importantly, inspection personnel will now inspect locations susceptible to degradation, as opposed to the Section XI inspection locations which are essentially randomly chosen, 2) the inspection techniques and qualification of personnel assure that the inspection personnel will be finely attuned to the mechanism of interest when conducting the examinations, and 3) the volumes inspected by the N560 Program are, on average, larger than typical Section XI examination volumes. In summary, implementation of the N560 Program will result in a safety improvement.

Even when conservative estimates of the impact of N560 are used (neglecting the positive impact of the N560 selection process and inspection for cause philosophy), the impact on risk is negligible. Calculated values are orders of magnitude below DG-1061 acceptance criteria of 1E-06/yr. Applications of this magnitude are considered risk neutral.

Principle #5: Performance-based implementation and monitoring strategies

The licensee's response to this principle falls into three parts. They are: 1) performance-based implementation, 2) Section XI required monitoring and feedback, and 3) other monitoring and feedback mechanisms. They are described as follows:

Performance-based Implementation

The basis for ASME Code Case N560 is the exceptional performance history of category B-J piping welds. A detailed data review of industry piping failures was conducted in support of the code case. In addition, a review of Vermont Yankee specific history was conducted. Although a number of instances of IGSCC were identified, these occurred prior to the piping replacement effort of the late 1980s. With that issue resolved, Vermont Yankee category B-J experience is consistent with the industry data.

On top of the exceptional performance, N560 provides a mechanism for identifying what locations to inspect and what techniques to use based upon the operating performance of like components.

Section XI Required Monitoring and Feedback

N560 requires that the existing monitoring and feedback mechanisms provided in Section XI be maintained. These are as follows:

- pressure and leak testing of all category B-J components,
- inspection results shall be compared to PSI and prior ISI (IWB-3130(c),
- for flaws exceeding acceptance criteria (IWB-3500),

  - increase the number of inspections to include those items scheduled for this and the next scheduled period (IWB-2430(a)),
  - additional inspections - all items of similar design, size and function (IWB-2430(b))
  - flaw - removed, repaired, replaced or analytical evaluation (IWB-3130/3140)
  - if accepted by analytical evaluation, items shall be examined for the next three inspection periods (IWB-2420(B))

Other monitoring and feedback mechanism

- Vermont Yankee Technical Specification 3.6.c

  - Unidentified Reactor Coolant Leakage shall not exceed 5 gpm
  - Total Reactor Coolant Leakage shall not exceed 25 gpm

- Feedwater Nozzle for bypass flow with (4) thermocouples per nozzle
- Drywell Monitoring

  - Radiation
  - Temperature
  - Pressure

Based upon the exceptional performance history of category B-J components, a detailed review of industry experience, and the multiple means of monitoring and providing feedback, N560 can be implemented in a streamlined yet robust manner."

3.0 NRC STAFF EVALUATION OF THE LICENSEE'S PROPOSED ALTERNATIVE

The NRC staff reviewed the licensee's request for approval of ASME Code Case N-560 to be used as an alternative to ASME Code, Section XI at the VY Nuclear Power Station. The Code Case provides alternative examination requirements to those stated in Table IWB-2500-1 of the ASME Code, Section XI and relates to examination category B-J, pressure retaining welds in piping. At the VY station, Code Case N-560 is augmented by the use of methodology developed by EPRI. In the course of its review, the staff transmitted requests for additional

information (RAIs) to the licensee and received licensee responses to these RAIs. In addition, several discussions were held between the staff and the licensee to clarify and resolve outstanding issues. Listed below are the results of the NRC staff review of the licensee's submittal.

3.1 Proposed Alternative Meets Current Regulations

The licensee stated that 10 CFR 50.55a and Appendix A to 10 CFR Part 50 are the primary regulations governing inservice inspection of piping. The intent of these regulations is to detect service induced degradation and assure a robust reactor coolant pressure boundary. 10 CFR 50.55a, refers to Section XI of the ASME B&PV Code which requires that Class 1 pressure retaining welds are periodically inservice inspected to ensure the integrity of the piping systems. The licensee also stated that the ASME Code, Section XI is the implementing vehicle for performing the Code required inspections and as such Code Case N-560 is an ASME approved alternative to current Section XI requirements.

Further, other Section XI inspection activities such as the examination of Class 1 socket welded connections and dissimilar metal welds, pressure and leak testing requirements, Class 2 and 3 piping examinations, are independent from the proposed change; consequently, these examination requirements are not affected by implementation of N-560. In addition to the other Section XI activities not affected by the implementation of N-560 are the augmented inspection programs, such as those made in response to Generic Letter 88-01 (NRC Position on IGSCC in BWR Austenitic SS piping) and Generic Letter 89-08 (Erosion/Corrosion Induced Wall Thinning).

The staff finds the licensee's reasoning to be valid and acceptable for use at the VY station because the ASME Code Case is a Code approved alternative and N-560 clearly states that its alternative requirement pertains only to examination category B-J welds and, therefore, the rest of the Code mandated requirements are not affected by the Code Case. Further, the licensee has augmented the Code Case by the use of methodology developed by EPRI and has adequately described in its transmittal how the Code Case N-560 will be applied at the VY station.

The results of the staff review of the proposed implementation of Case N-560 is also documented in this safety evaluation. The staff notes that the guidance in Code Case N-560 was augmented in certain areas; in the VY pilot application the EPRI methodology (EPRI TR-106706) was used as a basis for performing the degradation mechanism evaluation, the system impact group assessment, and containment performance impact assessment.

3.2 Maintain Defense in Depth

The licensee stated that the intent of the inspections mandated by ASME Section XI for category B-J piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in the reactor coolant pressure boundary. Currently, the

process for choosing inspection locations is based upon structural discontinuity and stress analysis results for plants licensed after July 1, 1978, or may be randomly selected for plants licensed prior to this date. The ASME Code evaluated the process of choosing inspection locations under the current rules of ASME Code Section XI and the results of the evaluation are documented in the ASME White Paper 9201-01, Revision 1, "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J, Pressure Retaining Welds in Piping." To provide alternative to the current methods of choosing inspection locations the ASME Code issued Code Case N-560, which uses a risk-informed approach to select inspection locations for ASME CODE, Section XI, Category B-J piping welds. The selection process adapted in the Case is a more robust selection process founded on actual service experience with nuclear plant piping failure data.

The ASME Code Case N-560 selection process has two key ingredients: (1) determination of each location's susceptibility to degradation and (2) an assessment of the consequence of the location's failure. These two ingredients not only assure defense in depth is maintained, but actually the defense in depth is enhanced, as compared to the current process. First, by evaluating a location's susceptibility to specific forms of degradation, the likelihood of finding flaws or indications that may be precursors to leaks or ruptures in the reactor coolant pressure boundary is increased. Secondly, the consequence assessment effort has a single failure criterion so that, no matter how unlikely a failure scenario is, it is ranked high if, as a result of the failure, there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability so that less reliable equipment is not credited as much as more reliable equipment.

Further, in the VY submittal the licensee stated that failure of each pipe section that results in significant loss of redundancy, is evaluated as having a "high" consequence rank independent of "frequency of challenge." That means that even for a very unlikely challenge, breaks that lead to loss of redundancy are evaluated as "high." For every pipe section which, if it breaks, results in an increase in probability of containment bypass is evaluated and, if the probability of bypass is significant, the assigned consequence rank is always "high." In the final selection, no significant reduction in inspection is recommended for all systems performing the same function.

The staff finds the licensee's approach acceptable because the weld selection process, as described by the licensee in its submittal, will ensure that welds which have a high failure consequence and are most susceptible to degradation will be targeted for examination. This process should result in the detection of flaws that challenge the pressure boundary of the subject piping and thus the integrity of the piping systems will be maintained.

3.3 Maintain Sufficient Safety Margins

The licensee concluded that attributes critical to defining and maintaining sufficient safety margins are unchanged, with the exception of certain pressure boundary volumetric examinations. In the VY case, augmented inspection programs will continue, and the reduced number of volumetric Section XI examinations are based upon the exceptional performance

history of category B-J components. In addition, the new inspection locations are more appropriate, usually involve larger inspection volumes, and improved examination techniques will be applied. Further, in anticipation of pending new ASME Section XI, Appendix VII and VIII requirements, VY is utilizing, in most cases, ultrasonic detection and sizing personnel who are qualified in accordance with the most current industry qualification programs. VY is the co-owner of a large selection of flawed ultrasonic examination qualification coupons that are used in conjunction with a performance-based qualification and certification program. Also, prior to implementation of the risk informed ISI program VY will develop training guidelines for the instruction of NDE personnel in the specific degradation mechanisms and enhanced examination areas associated with each mechanism. When the NRC formally mandates Appendix VIII of Section XI or approves a Section XI Code Edition containing Appendix VIII and the VY ISI program is updated to that Code Edition, Appendix VIII will be formally applied at the VY nuclear station. The licensee is required by ASME Section XI, Article IWA 1000, to file a revised ISI program with the NRC incorporating the requirements of Code Case N-560 and the licensee has committed to do so in its letter dated August 15, 1997. Further, VY does not rely on its NDE subcontractor to choose inspection personnel. The licensee selects NDE technicians based upon past demonstration of capability and, typically, the same technicians return to perform examinations at the VY station during each outage. VY looks for NDE examination processes and technicians that are qualified to meet the industry sponsored performance demonstration initiatives (PDI) even though Appendices VII and VIII have yet to be implemented at the VY station.

The NRC staff finds the VY qualification of NDE processes and personnel acceptable. Properly qualified NDE processes and personnel will ensure that unacceptable defects and flaws in piping will be reliably detected during the ISI examinations and, therefore, sufficient safety margins will be maintained in piping systems.

3.4 Proposed Increase In Risk Including Cumulative Effects Is Small; NRC Safety Goals Are Not Exceeded

The NRC staff reviewed the licensee's proposal to use Code Case N-560 as augmented by the use of EPRI methodology at the VY station in lieu of the requirements of ASME Code, Section XI, Table IWB-2500-1. The purpose of the review was to ascertain that the proposed increase in risk is small and that the NRC safety goals are not exceeded. Assurance that any proposed increase in risk is small is attained through the integrated analysis, evaluation, and decision making process followed by the licensee to develop the requested alternative to the ASME requirements. The results of the NRC staff review follows:

3.4.1 Scope of Piping Systems

The scope of the submittal encompasses all Class 1 examination Category B-J piping welds excluding socket welds.

Per article IXB-1220(a) of Section XI of the ASME Boiler and Pressure Vessel Code, components may be exempt from volumetric and surface examinations if the rupture flow

produced is within the capacity of makeup systems which are operable from on-site emergency power. For Vermont Yankee this results in pipe sizes of 2-inch NPS and less for water piping and 11/2-inch NPS and less for steam piping being excluded from volumetric and surface examinations.

3.4.2 Piping Segments

Pipe segments are defined as lengths of pipe whose failure lead to the same consequence and which are exposed to the same degradation mechanism. That is, some lengths of pipe whose failure would lead to the same consequences may be split into two or more segments when two or more regions are exposed to different degradation mechanism. The staff finds this appropriate, and necessary, because the methodology combines separate consequence categories with degradation mechanism categories and therefore the two characteristics should not be mixed within a segment.

3.4.3 Piping Failure Potential

In order to evaluate the potential for failure of the piping, the licensee identified the potential degradation mechanisms for each pipe segment within the scope of this proposal. Specifically, the following nine systems were included in the evaluation: core spray (CS), feedwater (FW), high pressure coolant injection (HPSI), main steam (MS), main steam drain (MSD), reactor core isolation cooling (RCIC), reactor water recirculation (RWRS), residual heat removal (RHR), and reactor water cleanup (RWCU) system. Each pipe segment was evaluated for the following sixteen degradation mechanisms: thermal fatigue (TF), thermal stratification, cycling, and striping (TASCS), thermal transients (TT), stress corrosion cracking (SCC), integranual stress corrosion cracking (IGSCC), transgranual stress corrosion cracking (TGSCC), external chloride stress corrosion cracking (ECSCC), primary water stress corrosion cracking (PWSCC), localized corrosion (LC), microbiologically influenced corrosion (MIC), pitting (PT), crevice corrosion (CC), flow sensitive (FS), erosion-cavitation (E-C), flow accelerated corrosion (FAC), and waterhammer (WH). Each degradation mechanism was evaluated for its potential impact on the pipe segment and each pipe segment was assigned a pipe failure potential based upon the degradation mechanism. Detailed information concerning pipe segments and pipe degradation evaluation is included in Table 1 and Table 2 of the licensee's August 15, 1997 submittal.

The NRC staff finds the licensee's evaluation of the potential for piping failure acceptable because the methodology provides for the evaluation of the pipe segment break consequence and incorporates consideration for potential degradation mechanisms.

3.4.4 Consequence of Failure

The consequence of the postulated pipe failure considered include both direct and indirect effects of each segment failures. The direct effects always include a diversion of flow large enough to either disable the system or lead to isolation (automatic if available, manual if feasible). Indirect effects include spacial effects caused by flooding, spray, and pipe whip as

well as depletion of water source such as draining tanks. The licensee stated that there are no spacial effects identified following pipe ruptures inside the drywell because of the environmental qualification and spacial separation of the equipment.

The analysis is performed assuming a large break based on the pipe diameter unless a smaller break results in more severe consequences. No credit is given to leak-before-break. The staff finds that the use of large break and all associated direct and indirect effects to bound the impact of each break can be used to characterize the risk from each segment and therefore, is acceptable.

In some cases, the equipment and functions lost as a result of a pipe ruptures can vary greatly if an automatic (e.g., check valve closure and automatic isolation valves) or manual isolation succeeds or fails. When large variations in lost equipment and functions are identified, the consequence category (which includes a measure of the likelihood of isolation) for both successful and unsuccessful isolation is developed and the highest category selected. The staff finds this process acceptable because it includes the systematic consideration of the potential for isolation failure, and, when applicable, considers the consequence and likelihood of isolation failure and isolation success.

A Failure Mode and Effects Analysis (FMEA) was performed for each segment rupture and the results for each segment included in the submittal. The FMEA was based on information from internal flooding study notebooks and site arrangement drawings. Key attributes collected during the flooding analysis walkdowns included flooding sources and flow rates, relief paths and openings to adjacent areas, location of equipment and height above floor, room penetrations, curbs, and overflows, and the potential for spray effecting equipment. Flood propagation paths are identified, described, and evaluated. The staff finds that the process described by the licensee, and as illustrated by the submitted results, indicates that the licensee's flooding analysis walkdowns were of sufficient scope and depth to ensure that the spacial effects evaluation is based on the current plant configuration. Furthermore, the staff finds the development of the spacial effects of pipe ruptures acceptable since appropriate information is collected, documented, and evaluated.

The licensee has considered pipe ruptures that only cause a transient, those that cause a transient and the loss of one or more mitigating plant trains or systems, and those that do not cause a transient but which fail one or more mitigating plant trains or systems. As discussed in 3.4.6, the large early release frequency (LERF) considerations are included by considering the conditional containment failure probability associated with the segment ruptures. The staff finds that all relevant consequences of pipe failures are included in the evaluation.

3.4.5 Probabilistic Risk Assessment

The Vermont Yankee Individual Plant Examination (IPE) was completed in December 1993. Excluding internal floods, the IPE estimated a core damage frequency (CDF) of 4.3E-6/yr. Defining LERF as greater then 10% iodine release within 6 hours of core damage, VY reported a LERF of 9.5E-7/yr. The first formal update to the VY IPE is planned for the end of 1998.

The internal flooding analysis was performed in the Individual Plant Examination of External Events (IPEEE) study instead of, as is usual, in the IPE study. The IPEEE was submitted in June 1998. The internal flooding CDF was estimated to be 9.0E-6/yr.

The licensee used quantitative PRA results to support the following evaluations.

a) The licensee uses PRA results to estimate the conditional core damage probability (CCDP) for segment failures which cause only initiating events (primarily loss of coolant accidents (LOCA)). The CCDP was obtained by dividing the CDF for each relevant initiating event by the initiating event frequency. This result can be directly compared to the CCDP guidelines to place the segment in the appropriate consequence category. The guidelines are discussed in Section 3.4.6.

b) For those segment failures which only fail mitigating systems and do not cause a plant trip, system unavailabilities were used to determine the equivalent number of "back-up trains" remaining available to mitigate independently occurring initiating events.

c) For those system failures which fail mitigating system(s) and which cause a plant trip the IPE was re-quantified to determine the CCDP.

The methodology assigns segments into consequence categories based on the probabilities of core damage and large early release given that each segment has failed. Although the PRA was used to develop CCDPs, it was not used to determine any large early release probabilities (LERP) values. LERP considerations were evaluated using a conditional containment failure potential which is estimated from the level II analyses in the PRA as necessary.

The staff finds that appropriate results from the IPE were used in a manner consistent with their definitions.

Quality of PRA

The IPE was performed under the direction of, and with much support from, an engineering services organization that provides dedicated support to the licensee. All aspects of the IPE analysis received a thorough review by qualified licensee persons who were "independent" to the extent that the reviewers were not directly involved in the analysis they reviewed. Comments developed during the reviews were incorporated into the final IPE. System models were reviewed during meetings attended by licensee representatives from systems engineering, electrical engineering, mechanical engineering, instrumentation and controls engineering and operations. Several external consultants supported human reliability, success criteria, and containment phenomenological evaluations.

The IPE review noted that the licensee's involvement in the IPE process was more substantial than in some other IPE studies and that the plant-specific treatment of some accident sequences indicated considerable involvement of knowledgeable utility staff.

The IPE used to support the RI-ISI submittal was the original 1993 study. The licensee reported that a review of collected equipment unavailability data (as opposed to the generic data used in the IPE) indicated that any changes would not apply to the RI-ISI relevant results. Changes to the plant design and procedures are saved in a book as input to the planned IPE update. The changes were reviewed by the licensee and judged to have no impact on the RI-ISI selections.

The IPEEE analysis, including the internal flooding, was performed by a team of licensee and consulting engineers having detailed knowledge of the Vermont Yankee systems design and arrangement and the IPE study. Review by an independent consultant firm found the analysis thorough and consistent with NRC guidance regarding internal floods and other external events. The reviewers generated a number of questions and comments which were resolved. The results discussed in the submittal indicate that the licensee identified equipment in the various areas which could be susceptible to the environmental impact of the pipe rupture, and the environmental qualification of the equipment.

The staff finds that the extensive involvement of the licensee's staff, the numerous reviews performed by the licensee, and the maintenance of the plant design and procedural change notebooks demonstrate the licensee's commitment to the quality of the IPE and the flooding analysis in the IPEEE. Earlier staff review of the IPE concluded that it met the intent of generic letter 88-20 and identified no particular shortcomings. The maintenance rule inspection found the quality of the IPE appropriate to perform the risk categorization in accordance with the maintenance rule. Based on the submittal and the results of the ISI evaluation reported by the licensee, those parts of the IPE used extensively to support the submittal were identified. A focused review of those parts of the IPE identified no shortcomings that might invalidate the results of the evaluation used to support the submittal.

The staff did not review the results of the IPE or the flooding analysis to assess the accuracy of the quantitative estimates. The staff recognizes that the quantitative results of the IPE are used as order of magnitude estimates for a variety of risk and reliability parameters. These estimates are used to support the assignment of segments into three broad consequence categories. Additional estimates are used to confirm parts of the methodology whereby segments are assigned to consequence categories based on the number of back-up trains available versus demand occurrence frequency categories. Inaccuracies in the models or assumptions large enough to invalidate the broad categorizations developed to support RI-ISI should have been identified in the licensee or the staff reviews. Minor errors or inappropriate assumptions will only affect the consequence categorization of a few segments and not invalidate the general results or conclusions. The staff finds the quality of the PRA sufficient to support the submittal.

Scope of PRA

The VY IPE completed in December 1993, did not include shutdown, fires, and seismic events. Special exemption (NVY 90-057) was given to VY to allow internal flooding to be included in the IPEEE instead of the IPE. VY submitted its internal flooding analysis with its IPEEE study in June 1998.

Segments in the high consequence category are not further differentiated, so any segment categorized as High in the internal event evaluation need not be further evaluated for other types of initiating events. The licensee evaluated the shutdown modes of operation and operating conditions for each of the systems included in the evaluation. Only the RHR systems was found to be potentially more important during shutdown. The detailed discussion in the submittal concluded that one RHR segment initially categorized low should be categorized medium.

Fires, external floods, and seismic initiating events are also evaluated and discussed in the submittal. As with shutdown, only pipe segments with medium and low consequence need to be reviewed. In general, the licensee determined that the frequency of transients related to class 1 piping and induced by external events is less than the frequency of internal events induced transients. No relationship between reduced inservice inspections and increased vulnerability to segment failure arising from the occurrence of external events was identified.

The staff finds the scope of the IPE acceptable because initiating events and operational modes outside of the scope of the IPE were systematically included in the evaluation and not neglected.

3.4.6 Safety-Significance Determination

The PRA is used to support the categorization of pipe segments' failure consequences into one of three broad categories, high, medium, or low. These results are coupled with degradation evaluation categories to support the more systematic and efficient selection of ASME Section Xi class 1 B-J pipe welds to inspect.

Quantitative uncertainty calculations are not included in the methodology. The placing of segments into broad safety significant categories tends to reduce the sensitivity of the eventual decision on the specific values developed from the PRA, with the exception of values near the border between the categories. The sensitivity of the values near the borders is addressed by defining a medium consequence category. The medium consequence category ensures that segments whose failures cause consequences which are not obviously high or low are treated as Medium (intermediate) severity segments, both during the final safety-significance determination and in the assignment of weld elements to inspect. The staff finds that the performance of quantitative uncertainty calculations would not provide information which would significantly change the results of the submittal.

Consequence Categorization

The specific decision criteria used to determine the consequence category depends on the type of impact the segment failure has on the plant. In general, however, the criteria are derived from guidelines applied to the CCDP given the segment failure. That is, given a segment failure and all the associated spacial effects, the CCDP is the probability that the resulting scenario will lead to core damage. If the failure of a segment is estimated to lead to a core damage event with a probability greater the 1E-4, the segment is categorized as High consequence. An estimated CCDP within the range of 1E-6 to 1E-4 is categorized as Medium consequence. CCDPs less then 1E-6 are categorized as Low consequences.

The methodology provides guidance on assigning consequence category to segment breaks based on the number of available trains, broad categories of initiating event frequencies, and exposure times. The licensee also explicitly developed order of magnitude CCDP estimates for each segment and compared the estimates to the above CCDP guidelines. The results from the two methods are dependent insofar as the quantitative estimates are used to determine the equivalent number of back up trains. Nevertheless, the staff finds that the parallel evaluations provides additional confidence that each segment is assigned an appropriate consequence category.

The following decision criteria are used to support the CCDP related categorization of each type of segment failure consequence.

a) When the segment failure causes only an initiating event (e.g., no mitigating system failures caused by segment rupture) the CCDP can be estimated and directly compared to the guideline values. Most class 1 pipe failure leads to LOCA accidents, so most segment failures in this submittal were categorized with this method.

b) Segment failures which only fail mitigating functions but do not cause a plant trip increase the likelihood that, following an unrelated initiating event, the sequence of events will lead to a core damage event. In some cases (for example, normally isolated segments) the segment failure may occur before the event but only become manifest upon demand. In other cases, the failure may be detected and repair initiated (up to the allowed outage time limits of the equipment) and the event occur during the repair. The licensee uses a matrix supplied in the submittal that specifies consequence categories based on categories of initiating events based on expected frequencies, the number of equivalent, unaffected trains left to mitigate the event, and exposure time. The specified consequence for each matrix entry was developed by developing a CCDP from the bounding values of all three contributors, and comparing that bounding value to the CCDP guidelines.

c) Segments which both cause an initiating event and fail mitigating systems are the last type of segment failure consequences. The licensee used a matrix supplied in the submittal whereby the number of equivalent, unaffected trains available for mitigation determines the consequence.

The matrices used in b) and c) require that each unaffected train left to mitigate an event has an unavailability of 0.01. That is, in order for the CCDP of the matrix elements assigned High, Medium, and Low to correspond to the 10E-4, between 10E-4 and 10E-6, and less than 10E-6 guidelines, each unaffected train must provide an availability of at least 0.99. Due to potential interactions between the system trains and between different systems, the licensee performed PRA calculations where the reliability of different combinations of trains was calculated and the number of equivalent backup trains determined. For example, the low pressure core injection (LPCI) mode of the residual heat removal (RHR) system has two separate flow loops and each loop has two pumps. The reliability of LPCI injection function is calculated to be 2.5E-4. Therefore, when none of the equipment associated with the LPCI function is directly impacted by the failed segment, the presence of 1.5 backup trains is credited (2 trains requires a maximum unavailability of 1E-4). The licensee also performed a number of PRA calculations verifying that interactions between systems and initiating events were appropriately considered. The staff finds the definition and use of equivalent backup trains acceptable because the licensee performed calculations using the base-line PRA to confirm the availability of the different system/train configurations.

The licensee's LERP related guidelines are a factor of 10 lower than the CCDP guidelines. (e.g., High > 1E-5 LERP, Low < 1E-7 LERP, and Medium otherwise.) The segment is assigned the higher of the CCDP or LERP consequence category. For segment failures which are deemed to by-pass containment (generally a segment failure and an isolation failure) the LERP guideline can be directly applied. That is, although a segment failure with a CCDP of 5E-5 with containment bypass belongs in the Medium category according to the CCDP guidelines, it is assigned High safety significant consequence based on the LERP guideline. Segments which do not directly bypass containment are also reviewed for LERP considerations. Segments with CCDPs greater then 1E-5 are reviewed and if the containment might be challenged, the level II analyses in the IPE are reviewed to determine the conditional containment failure probability. If the conditional containment probability yields a LERP greater then 1E-5, the consequence category is increased. Similarly, for CCDP scenarios between 1E-6 and 1E-7, the possibility of subsequent containment failure is investigated to determine if the consequence category should be raised from Low to Medium.

The staff finds that the licensee systematically considered all relevant types of segment failure consequences. The licensee also performed a number of confirmatory calculations with the base-line PRA models to ensure that system interactions were investigated and appropriately included. LERF considerations were included by reviewing scenarios where the loss of containment integrity might result in exceeding the licensee's LERP guidelines. If it could not be determined with confidence that the LERP guidelines would not be exceeded, the safety significance of the segment was increased.

The staff finds the consequence categorization process as applied by the licensee to be reasonable. The order of magnitude of the High consequence (e.g., severe consequences to receive substantial attention) guidelines are consistent with the CDF and LERF decision criteria that, if exceeded, may require the licensee to present arguments as to why steps should not be taken to reduce the CDF and LERF. That is, a plant continuously operating near the decision

criteria of a CDF of 1E-4/yr or a LERF of 1E-5/yr would, during the next year, have a CDP and LERP corresponding to the guidelines used by the licensee to define High consequence. The two orders of magnitude between the High and Low guidelines provide a robust Medium category such that it can be reasonably concluded that segments categorized as Low consequences make a negligible contribution to risk. Therefore, the staff finds that the licensee's guidelines are consistent with acceptable risk related guidelines and that they provide reasonable assurance that the segments are appropriately characterized.

Safety Significance Categorization of Pipe Segments

The safety significance of pipe segments is based on categorizing a) the consequence of segment failure into High, Medium, or Low; and b) categorizing the failure potential of the piping as High, Medium, or Low is discussed in Section 3.4.3. Once the individual elements of risk (consequence and failure potential) are developed, they are combined in a matrix that has nine elements, corresponding to various combinations of failure potential and consequence rankings.

These combinations define the basis for categorizing the pipe segments into various risk categories 1 through 7. Risk categories 1, 2, and 3 are designated as belonging to the High risk group, risk categories 4 and 5 belong to the Medium risk group, and risk categories 6 and 7 belong to the Low risk group. Examination zones are then selected by starting with the structural elements in the High risk group and working toward the Low risk group, until a total number of structural elements equal to 10% of the Category B-J piping welds, excluding socket welds, have been selected.

The Medium safety significance ensures that segments which are not clearly High or Low, will receive an intermediate level of inspection activity. The staff finds that the assignment of the safety significance to the nine matrix elements as detailed in the submittal is internally consistent and logically compelling. The staff finds that the use of the reported categories, along with other evaluation and confirmation steps detailed in this safety evaluation, provides reasonable assurance that the safety significance of each segment is appropriately assigned.

3.4.7 Determination of the Change in Risk

The VY submittal proposes to reduce the examination of ASME Code, Section XI welds from 113 welds to 41 welds (e.g., from 28% to 10%). The licensee evaluated the impact on CDF which might be associated with the change in the ISI program. The licensee's estimated a net CDF decrease on the order of -6E-10/yr when credit for improved inspection techniques they will apply as part of the new inspection program is included. When no credit for the improved inspections is included, the licensee estimates a CDF increase on the order of 3E-9/yr.

The licensee did not calculate the change in LERP. LERP was included in the analysis through the systematic identification of containment failure scenarios. In the seven High safety significance segments based primarily on LERP (e.g., segments with the potential for an interfacing system LOCA or a LOCA outside containment), the licensee proposes increasing

the number of inspections from 3 to 4. The number of locations in the Medium safety significance category segments will be reduced from 4 to 1, and the 3 inspections in the Low safety significance category will be discontinued. However, it should be noted that the pressure and leak testing requirements mandated by ASME Code, Section XI will still be performed on all welds because those requirements are not affected by the implementation of Case N-560. The staff finds that further estimation on change in LERF for so few changes in inspections would not contribute significantly to the assurance that any change in risk would be negative or minimal.

The staff finds the licensee's process to evaluate and bound the potential change in risk acceptable because it accounts for the change in the number of elements inspected, recognizes the difference in degradation mechanism related to failure likelihood, and considers the effects of enhanced inspection. The staff finds that the improved inspection techniques will substantially increase the fraction of potential weld failure which would be identified by the inspection before the flaw develops into an actual failure. The staff also finds that re-distributing the welds to be inspected with consideration of the safety significance of the segments provides assurance that segments whose failure have a significant impact on plant risk receive an acceptable level of inspection. Therefore, the staff concludes that the implementation of the RI-ISI program as described in the application will reduce, or negligibly increase risk, and thus will not cause the NRC safety goals to be exceeded.

3.4.8 Integrated Decisionmaking

Vermont Yankee performed an independent review of the safety significance analysis done to support the RI-ISI program. The analysis and subsequent review was carried out under the NRC staff approved 10 CFR 50 Appendix B Quality Assurance Program (YOQAP-1-A) in accordance with procedures that provide for adequate verification of the quality of the technical analysis and documentation of the performance of the verification. During the course of the review and approval, the licensee stated that there are numerous discussions on a variety of topics and concerns leading to eventual consensus that the results represent the best available information. The licensee provided signed title pages and review process checklists as part of the submittal, and maintains copies of the applicable procedures and referenced RI-ISI calculations available for staff review.

The final selection of welds is performed during a formal meeting and discussion. Although not established as a standing group, the panel meeting results are documented in an internal licensee memo. Based on the discussions on the safety significance of the segments (including an explanation on the consequence and the degradation mechanism responsible for placing the segment in the assigned category) the actual welds to be included in the RI-ISI program are selected.

The staff finds the review of the evaluations and results used to support this RI-ISI program change acceptable since, as described by the licensee, they provided for an independent review by personnel technically knowledgeable in the applicable engineering disciplines, and developed records of the review to be maintained for subsequent licensee re-evaluations and

staff review if necessary. The process used to select the specific welds to inspect is believed to results in a credible and supportable selection since the appropriate personnel make the final decisions based on a discussion of the information needed to support the decision. Although the licensee has not developed a formal, standing expert panel, the meeting where the final decisions are made has the necessary attributes (e.g., appropriate personal, authority to make decisions, and documented decisions) of a formal expert panel.

3.4.9 Selection of Examination Locations and Methods

The licensee used guidance provided in EPRI Report TR-106706, *Risk-Informed Inservice Inspection Evaluation Procedure*, for assessing potential degradation mechanisms, expected locations, and appropriate examination methods based on a review of the operating characteristics and plant experiences for piping systems at VY. Three types of degradation mechanisms were identified; 1) intergranular stress corrosion cracking (IGSCC), 2) flow-accelerated corrosion (FAC), and 3) thermal fatigue. For each of these mechanisms, specific guidance is provided in the EPRI procedure for choosing susceptible locations and applying volumetric examination methods to detect associated degradation. The licensee accounts for actual design features in VY plant piping, reviews accessibility for examination and considers minimizing personnel radiation exposure, then determines the optimum locations for inspection.

The staff finds that the guidance for selecting examination locations, descriptions of affected examination areas for mechanisms appropriate to VY, and prescribed volumetric methods provided in EPRI Report TR-106706, if followed, will provide reasonable assurance of the structural integrity of Class 1 piping systems. The licensee has committed to following this guidance, therefore, the staff finds the licensee's selection of examination locations and applied volumetric methods acceptable.

3.4.10 Conclusion

The staff finds that the licensee has developed an acceptable process to determine the relative safety significance of pipe segments and to select examination locations and methods for those locations. The integrated process identifies locations where degradation mechanism are more likely to occur and adapts examinations to those mechanisms and therefore represents an improved inspection strategy compared to the current strategy. The licensee also provides quantitative estimates supporting the expectation that the proposed alternative represents a risk decrease or, at most, a negligible increase which can be considered risk neutral. Since there is no proposed increase in risk, the application is consistent with Principle 4, "Proposed Increase In Risk Including Cumulative Effects Is Small; NRC Safety Goals Are Not Exceeded."

3.5 Implementation and Monitoring

The N-560 Code Case methodology, as applied to Vermont Yankee, constitutes a RI-ISI program that upon approval, would be implemented as the alternative to current Section XI requirements for Class 1 system B-J Category welds. As noted by the licensee in their submittal, any new risk insights, plant changes, or industry information that would have a

significant impact on the inservice inspection program, or the basis for NRC's approval of the program, would require a re-evaluation and any changes to the RI-ISI program would not take place without NRC approval.

The licensee has indicated as noted previously in Section 2.3, Principle #5, that existing monitoring and feedback programs provided in Section XI will be maintained including pressure and leak tests of all category B-J components, inspection results will be compared to PSI and prior ISI (IWB-3130c), and use of IWB-3500 will be followed for flaws that exceed acceptance criteria.

The licensee also stated that the results of NDE inspections are evaluated upon completion and discrepancies are documented per in-house procedures. Intermediate and long-term corrective actions are identified and tracked to completion. Corrective actions may include repairs, supplemental inspections, and a review of program adequacy. The goal of the program is to identify where the inservice inspection program needs to be revised or improved or when issues need to be assessed in order to verify structural integrity of the safety class systems.

The staff therefore concludes that performance-based implementation and monitoring strategies at Vermont Yankee will ensure that an acceptable level of quality and safety will be maintained.

4.0 CONCLUSION

The NRC staff finds that the licensee has provided an acceptable alternative to the requirements of the ASME Code Section XI. The licensee has shown that implementation of the program would result in an insignificant change in risk even with fewer inspections, since the inspections will take place where degradation mechanisms are more likely to occur, and procedures and personnel will target these specific locations using improved techniques and expanded volumes. The staff has determined that the alternative method described in the licensee's submittal (ASME Code Case N-560 as augmented by EPRI TR-106706) provides equivalent or better examination criteria for Class 1 Category B-J welds than that provided by the current Section XI requirements.

Therefore, the staff concludes that authorization of the licensee's proposed alternative would provide an acceptable level of quality and safety, in that the alternative provides reasonable assurance of the structural integrity of affected components. Pursuant to 10 CFR 50.55a(a)(3)(i) the alternative is authorized. This authorization does not constitute an NRC approval of Code Case N-560 for generic use. The suitability of Code Case N-560 for generic use will be determined following the staff's review of the Case. It is expected that the results of the staff's review will be documented in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI Division 1."

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