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Regulatory Analysis for the Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

T. Y. Chang

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ABSTRACT

Generic Safety Issue (GSI)-29 deals with staff concerns about public risk due to degradation or failure of safety-related bolting in nuclear power plants.

The issue was initiated in November 1982. Value-impact studies of a mandatory program on safety-related bolting for operating plants were inconclusive; therefore, additional regulatory requirements for operating plants could not be justified in accordance with provisions of 10 CFR 50.109. In addition, based on (1) operating experience with bolting in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry-proposed actions, the staff concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic failure of a reactor coolant pressure boundary joint that will lead to significant accident sequences is highly unlikely. For future plants, it was concluded that a new Standard Review Plan section should be developed to codify existing bolting requirements and industry-developed initiatives.

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EXECUTIVE SUMMARY

Generic Safety Issue (GSI)-29 was established in November 1982 to deal with staff concerns over the increase in the number of events involving the degradation of threaded fasteners (bolt cracking, corrosion, failure, etc.) reported by licensees of operating reactors from 1964 to the early 1980s. Many of the events were related to components of the reactor coolant pressure boundary (RCPB) and support structures of major components. This raised questions regarding the integrity of the RCPB and the reliability of the component support structures following a loss-of-coolant accident (LOCA) or a seismic event.

The potential safety significance of GSI-29 primarily involves the critical bolting applications that constitute an integral part of the RCPB, including closure studs or bolts on reactor vessels, pressurizers, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and jeopardize safe operation of the plant. Bolting is also an integral part of the pressure boundary and component supports of other systems, both safety-related and not. These and other bolting applications are essential for withstanding transient loads created during abnormal or accident conditions.

Failures reported by licensees involved a variety of threaded fasteners and several causes. As a result, several different failure mechanisms had to be considered. Most frequent were wastage (corrosion/erosion) from boric acid attack and stress corrosion cracking (SCC). The former occurred more often at RCPB joints, the latter in structural bolting.

The staff has concluded that a sufficient technical basis now exists for the resolution of GSI-29. This is based on the content and availability of material developed by industry and the NRC and the actions taken by both to address specific bolting problems observed in nuclear power plants.

Industry actions include the program developed by the Joint Atomic Industrial Forum/Materials Properties Council/Electric Power Research Institute (AIF/MPC/EPRI) Task Group on Bolting. Industry representatives established the Bolting Technology Council (an MPC affiliate) to take the lead in sponsoring bolting research, recommending practices, gathering and providing information, and promoting education on installation, application, behavior, and interactions of fasteners. The Institute of Nuclear Power Operations (INPO) has also taken a number of actions over the years in response to potentially unsafe conditions of degraded bolting and bolting-related issues. Further refinements in codes and standards are under way by the responsible committees in the American Society of Mechanical Engineers and the American Society for Testing and Materials.

Since 1982, the NRC addressed a number of specific bolting issues and took several additional steps that were considered in the resolution of the issue. The NRC has issued three bulletins, two generic letters, and eight information notices dealing with specific bolting problems. Many of the generic letters and bulletins required licensees not only to take short-term actions to resolve the problem but also to develop and implement continuing programs to minimize the likelihood of recurrence.

Value-impact studies of a mandatory program on safety-related bolting for operating plants were inconclusive; therefore, additional regulatory requirements for operating plants could not be justified in accordance with the provisions of 10 CFR 50.109. In addition, based on (1) operating experience with bolting in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry-proposed actions, the staff concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic failure of RCPB joints that will lead to significant accident sequences is highly unlikely.

For future plants, it was concluded that a new Standard Review Plan section should be developed to codify existing bolting requirements and industry-developed initiatives, including the development and implementation of a plant-specific bolting integrity program.

1. STATEMENT OF THE PROBLEM

The NRC observed that the number of events involving the degradation of threaded fasteners (bolt cracking, corrosion, failure, etc.) reported by licensees of operating reactors had increased from 1964 to the early 1980s. Many of the events were related to components of the reactor coolant pressure boundary (RCPB) and support structures of major components. This caused an increasing concern regarding the integrity of the RCPB and the reliability of the component support structures following a loss-of-coolant accident (LOCA) or a seismic event.

Originally an integral part of USI A-12, "Fracture Toughness of steam Generator and Reactor Coolant Pump Supports," the bolting safety issue was separately identified as Generic Safety Issue (GSI)-29, "Bolting Degradation or Failure in Nuclear Power Plants." The technical reason for that action was that the types and variety of failure mechanisms active in bolting safety problems were distinctly different from those to be addressed in structural steel supports. Bolting in the context of GSI-29 includes bolts, studs, embedments, cap/machine screws, other special threaded fasteners, and all their associated washers and nuts.*

When the NRC prioritized generic issues in November 1982, GSI-29 received a High rating. The safety aspects of GSI-29 primarily involve the critical bolting applications that constitute an integral part of the RCPB, including closure studs or bolts on reactor vessels, pressurizers, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and jeopardize safe operation of the plant. Bolting is also an integral part of the pressure boundary and component supports of other systems, both safety-related and not. These and other bolting applications are essential for withstanding transient loads created during abnormal or accident conditions.

Failures reported by licensees involved a variety of threaded fasteners and several causes. As a result, several different failure mechanisms had to be considered. Most frequent were wastage (corrosion/erosion) from boric acid attack and stress corrosion cracking (SCC). The former occurred more often at RCPB joints, the latter in structural bolting. Details regarding the nature and extent of bolting degradation and failure, including a review of the relevant available literature, can be found in NUREG-1339, "Resolution of Generic Safety Issue-29: Bolting Degradation or Failure in Nuclear Power Plants" (Ref. 1).

* Concerns regarding reactor vessel closure studs are being addressed under Generic Safety Issue-109, "Reactor Vessel Closure Failure," and therefore are not considered under GSI-29.

2. SUMMARY AND CONCLUSIONS

The conclusion that GSI-29 can be resolved is based on the content and availability of material developed by industry and the NRC and the actions taken by both to address specific bolting problems observed in nuclear power plants.

Industry actions include the program developed by the Joint Atomic Industrial Forum/Materials Properties Council/Electric Power Research Institute (AIF/MPC/EPRI) Task Group on Bolting. This effort resulted in two volumes of EPRI NP-5769 (Ref. 2), three video training tapes, and two volumes of the EPRI Good Bolting Practices Reference Manual (Ref. 3). Industry representatives established the Bolting Technology Council (an MPC affiliate) to take the lead in sponsoring bolting research; recommending practices; gathering and providing information; and promoting education on installation, application, behavior, and interactions of fasteners. The Institute of Nuclear Power Operations (INPO) has also taken a number of actions over the years in response to potentially unsafe conditions of degraded bolting and bolting-related issues. Further refinements in codes and standards are under way by the responsible committees in the American Society of Mechanical Engineers (ASME) and the American Society for Testing and Materials (ASTM). All of these industry actions and their impacts on GSI-29 are discussed in Reference 1.

Since 1982, the NRC addressed a number of specific bolting issues and took several additional steps that were considered in the resolution of the issue. The NRC has issued three bulletins, two generic letters, and eight information notices dealing with specific bolting problems as listed in Reference 1. Many of the generic letters and bulletins required licensees not only to take short-term actions to resolve the problem but also to develop and implement continuing programs to minimize the likelihood of recurrence. Details can be found in Reference 1.

Although value-impact studies on GSI-29 were performed by our contractors (Appendices A and B of this Regulatory Analysis), the staff judged the studies of a mandatory program on safety-related bolting for operating plants to be inconclusive; therefore, additional requirements for operating plants could not be justified in accordance with the provisions of 10 CFR 50.109. In addition, based on (1) operating experience with bolting in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry-proposed actions, the staff concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic failure of RCPB joints that will lead to significant accident sequences is highly unlikely. More detailed discussion of these considerations is provided in Sections 5.1 and 5.2. For future plants, however, it was concluded that a new Standard Review Plan (SRP) section should be

developed to codify existing bolting requirements and industry-developed initiatives, including the development and implementation of a plant-specific bolting integrity program.

3. **OBJECTIVE**

The objective of the proposed resolution of GSI-29 is to provide assurance that the integrity of safety-related threaded fasteners is maintained.

4. **ALTERNATIVES**

Several possible alternatives for the resolution of GSI-29 have been considered by the staff:

4.1 **No Action**

Take no actions beyond those already covered by existing NRC regulations, NRC bulletins/information notices and generic letters, and the ASME Code.

4.2. **Generic Letter**

Issue a generic letter (for information only) to owners of plants that currently have an operating license (OL) or a construction permit (CP) that suggests but does not require certain actions. The suggested actions include (1) a review of relevant industry-developed information and NRC documents and (2) development and implementation of a plant-specific bolting integrity program applicable to all safety-related joints.

4.3. **Guidance for Future Plants**

Develop a new SRP section to provide guidance to the staff for the review of future plants. The elements of the review would include the design of all safety-related joints, the selection of threaded fastener materials, and programmatic aspects dealing with bolting integrity during construction and operation/maintenance.

4.4 **Mandatory Program**

Require that plant owners currently holding an OL or CP (1) perform an engineering evaluation (or reevaluation) of all existing safety-related joint designs, fastener materials, and field practices (construction and maintenance) and (2) replace bolting (with redesigned joints, if necessary) that does not meet the industry-developed criteria or NRC requirements.

5. EVALUATION OF ALTERNATIVES AND DECISION RATIONALE

5.1. Costs and Benefits

Alternative 4.1 (taking no further action) was rejected because, although many requirements and much guidance have been published for bolting, no overall staff conclusions on the adequacy or effectiveness of these actions have been issued. As will be discussed later, alternative 4.4 (mandatory program), although clearly sufficient to ensure the integrity of bolted connections, is not warranted on the basis of the observed failures to date and the high cost of implementation. Therefore, Alternatives 4.2 and 4.3 were selected. Alternative 4.2 applies to plants that currently have a CP or OL, and Alternative 4.3 applies to future plants. The basis for selecting these alternatives is described in more detail in the following paragraphs. In summary, it was the judgment of the staff that existing ASME and ASTM codes and standards, NRC requirements and licensee actions (including bulletins and generic letters), and information available from the industry (e.g., Ref. 2) and NRC reports in the NUREG and NUREG/CR series would adequately limit the risk resulting from safety-related bolting failure in current plants. However, it was decided to inform licensees of the staff conclusions regarding the adequacy or effectiveness of the above-mentioned bolting-related requirements, guidance, information, and activities and to suggest but not require that current licensees develop and implement a plant-specific overall bolting integrity program that includes current NRC requirements and reflects the information and recommendations made by the industry-sponsored program. Such a plant-specific bolting integrity program should incorporate licensee commitments for continuing actions made in response to the previously issued NRC bulletins and generic letters listed in Reference 1.

New plants would be reviewed in accordance with a new Standard Review Plan section that would codify existing guidance and industry-developed recommendations. This guidance is justified for future plants because this represents only a codifying of existing guidance and practices.

The selection of these alternatives was based partially on two value-impact analyses: Appendix A on RCPB bolting and Appendix B on safety-related bolting in systems other than the RCPB. The staff's judgment of uncertainties and the impact of on-going activities that are not reflected in the value-impact analyses were also major factors in the decision-making process.

5.1.1 RCPB Bolting

Regarding the Pacific Northwest Laboratory (PNL) value-impact analysis of RCPB bolting (Appendix A), the best estimate indicated that the action proposed by PNL had the potential to reduce risk

by 9,819 person-rem for the whole industry. This was based on a best estimate of a reduction in core damage frequency of $2.73\text{E-}6$ /reactor-year for pressurized water reactors (PWRs) and $2.9\text{E-}7$ /reactor-year for boiling water reactors (BWRs). This magnitude of risk reduction is not considered by the staff to satisfy the 10 CFR 50.109 criterion that a required action must result in a substantial increase in the overall protection of the public health and safety. Further, in the staff's opinion, these estimated values of risk reduction probably erred on the high side. Considering the operating experience with bolting in both nuclear and conventional power plants (see Section 5.2), the actions (through bulletins, generic letters, and information notices) already taken since Reference 1 was prepared, and the industry-proposed actions, the staff concluded that leakage of bolted pressure joints is possible but catastrophic RCPB joint failure that will lead to significant accident sequences is highly unlikely.

The PNL value-impact analysis, however, did result in a best-estimate cost-benefit ratio of \$239 per person-rem, which is very favorable. When cost savings in public and onsite property damage are considered (these values, in the staff's opinion, have very large uncertainties), the cost-benefit ratio is even more favorable. As a matter of fact, because of the potential cost savings, the cost-benefit ratio turned out to be negative.

The best-estimate analysis was based on the assumption that all carbon or low-alloy steel bolts would be susceptible to boric acid wastage and would be replaced by stainless steel bolts. Such a program would be quite expensive for plants already constructed, and the staff believes that the PNL study underestimated the cost. Furthermore, an extensive program for inspecting and replacing RCPB bolting (beyond that required by Section XI of the ASME code and the requirements of IE Bulletin 82-02) might require an increase in the duration of refueling outages. These costs were not included in the PNL study.

If more realistic cost estimates are employed, an increased cost-benefit ratio would result. In the staff's judgment, a more realistic estimate of the cost-benefit ratio would exceed \$1000/person-rem for plants currently holding an OL or CP.

The staff therefore concluded that a mandatory replacement program for RCPB bolting could not be justified for plants that currently have an OL or CP, and Alternative 4.2 was selected instead. However, the staff further concluded that the PNL value-impact analysis was more valid for future plants. It should be noted, however, that the staff does not specifically recommend in the proposed SRP that only stainless steel should be used for RCPB bolting. Bolting material should be selected after careful consideration of all of the concerns addressed in the NRC and EPRI publications discussed in the generic letter. The SRP section

proposed by the staff for future plants includes provisions for a comprehensive bolting integrity program that deals with initial design, material selection, and construction and maintenance practices. Therefore, Alternative 4.3 was selected for future plants.

5.1.2 Non-RCPB Bolting

The Idaho National Engineering Laboratory (INEL) study (Appendix B) examined the risks related to failure of safety-related bolting in systems other than the RCPB. Approximately ten safety systems were examined for risk sensitivity. In addition, the primary coolant system component supports also were examined in the risk analysis. The risk analysis was based on a hybrid probabilistic risk assessment (PRA) model developed after review of six plant-specific PWR PRA models. Based on this hybrid PRA model, it was concluded that the most significant risk associated with degraded bolting was the failure of either the on-site emergency power system (including associated support systems) or the reactor coolant system (RCS) component supports during a severe seismic event. Although the analysis was based on PWR plants, it is the staff's judgment that the results are also generally applicable to BWRs since the risk is dominated by seismic considerations.

The INEL best estimate of core damage frequency was $3.5\text{E-}5$ /reactor year and the corresponding public risk reduction was 7300 person-rem based on 67 operating PWRs. The corresponding cost-benefit ratio reported by INEL was \$3700/person-rem. This cost-benefit ratio excluded consideration of on-site property damage and averted occupational radiation exposure. When INEL included this consideration, the result was a net cost saving.

The INEL study has the following uncertainties:

1. Calculation of Reduction In Core Damage Frequency

As stated above, the INEL risk study was based largely on seismic risk. INEL calculated that there was a significant increase in the risk frequency of core damage that would result from severe seismic events if RCS component supports or the emergency power system anchorages were degraded. The INEL study assumed that all of this risk was associated with degraded threaded fasteners.

Since RCS component supports and equipment anchorages consist of more than threaded fasteners (e.g., welded anchorages for electrical cabinets), this is clearly an overestimate of the benefit that could be achieved by surveying and testing threaded fasteners and replacing those found to be degraded.

Moreover, the risk contribution from vibratory equipment such as pumps and air compressors probably was overestimated. Since they are subjected to vibratory loading and stress under normal operating conditions, degraded bolts will be uncovered during normal operation or during normal maintenance and inspection.

2. Calculation of Collective Dose

The INEL calculation of the public exposure resulting from core damage was based on the Seabrook plant, which was one of the six plants used for constructing the hybrid PRA model. The Seabrook containment is more robust than that of a typical PWR. As a result, the collective dose (person-rem) calculated was relatively low considering the high estimate of core damage frequency. In this regard, the staff's judgment is that the public exposure would be higher than the results from the INEL analysis for a typical operating plant if the core damage frequency were as high as that calculated by INEL.

3. Cost Estimates

The cost estimates in the INEL study probably were too low if the proposed sequential steps of surveying, testing, and replacing were carried out. These steps are likely to require an extensive bolting inspection and replacement program and might require an increase in the duration of refueling outages, which was not included in the cost analysis.

Considering the uncertainties cited above, the staff judged the INEL proposed program to be marginal for plants currently holding an OL or CP from the viewpoints of reduction in risk to the public and cost-benefit ratio. The staff therefore concluded that requiring a program such as the one proposed by INEL for safety-related bolting other than RCPB applications could not be justified for plants that currently hold an OL or CP, and Alternative 4.2 was selected instead. However, the staff further concluded that, for future plants, a more effective review as delineated in a proposed new SRP section should be followed.

The elements of the proposed SRP section would include the design of all safety-related joints, the selection of threaded fastener materials, and programmatic aspects dealing with bolting integrity during construction and operation/maintenance. When the proposed SRP section is implemented during new construction, essentially all the potential risk reduction associated with safety-related bolting can be readily achieved. The staff believes that the risk reduction and costs presented in Appendices A and B should be more reasonably applicable for a new plant. Also, the staff concern that the cost estimates did not consider

the impact on the duration of outages would not apply to new plants.

Alternative 4.4, although clearly sufficient to ensure the integrity of bolted connections, is not warranted on the basis of the observed failures to date and the high cost of implementation. It is judged that only a small additional reduction in risk would be achieved by Alternative 4.4 relative to Alternative 4.2 for plants that currently have an OL or CP. However, the engineering, labor, and material costs for Alternative 4.4 would be considerably higher than those for Alternative 4.2. More important from a cost viewpoint, some additional outage time would be needed to replace suspect bolting, even if that work were performed during planned refueling outages. The combination of replacement power costs and higher labor and material costs would result in a cost-benefit ratio that is much less favorable than that of Alternative 4.2. Therefore, Alternative 4.4 was not selected.

5.2. Operating Experience

The inconclusive nature of the contractors' value-impact analyses on GSI-29 regarding a mandatory program on safety-related bolting for operating plants discussed in Section 5.1 prompted the staff to look into operating experience with bolting in nuclear and conventional power plants, especially in pressure boundary applications. The findings are summarized in the following sections.

5.2.1 Experience With Pressure Boundary Applications

During a meeting of the ACRS Subcommittee on Materials and Metallurgy on January 9, 1991, the staff's proposed resolution of GSI-29 was reviewed. The transcript of the meeting (Ref. 4) contains the following statement by John Bickford, past chairman of the Bolting Research Council, on his twenty years of operating experience with pressure boundary bolting:

"The thing that we were concerned about [in a nuclear power plant].... was radiation released which might be caused by a large or small LOCA or damage to components which would prevent a smooth shutdown in case of an emergency or just in general.

".... None of those things had been actually reported. we were generating this information from safety-related reports from the operating plants.

"These things as far as our committee work was concerned, LOCAs and so forth, might have been caused by either simultaneous failure of several bolts -- in other words a joint failure, unzipping as has been talked about, or loose

parts in the system and those things might be preceded by the rupture of individual bolts or the loss of individual bolts.

"Now loose parts in the system had been observed and were reported. Rupture of individual bolts had been observed and reported. Loss of individual bolts had been reported. Simultaneous joint failure had not been reported. [Emphasis added.]

"I think it might be pertinent to say that [I am aware of a] Tampa Electric Company [incidence which involved] the total failure of a [pressure] joint. I believe it was in a heat exchanger in a conventional power plant.

The problem was that the joint had been sealed with Furmanite [sic], which had trapped corrosive materials and so forth inside this thing and the joint just suddenly exploded and one person I believe was killed. That is the only incident that I am aware of in 20 years of bolting where a pressure vessel joint has failed catastrophically like that. [Emphasis added.]

"Many times leaks, many times partial failures, but never -- that's the only incident I know of, of that kind."

It is worthwhile to note from the above presentation that, to date, no LOCA, large or small, due to failure of bolted RCPB joints has been reported in any of the operating nuclear plants. The only "unzipping" type failure of a pressure boundary of any kind in any industry, based on J. Bickfords' extensive experience with bolting, was in a heat exchanger at a conventional power plant, and this was due to the use of Furmanite sealant. In NRC IE Bulletin 82-02, the licensees were cautioned on the prolonged use of improper sealants, and advised on the proper selection, procurement, and application of fastener sealant compounds to minimize the susceptibility of fasteners to SCC environments.

5.2.2 Comparison of Estimate From Contractor's Value-Impact Report With Operating Experience

The PNL value-impact analysis (Appendix A) estimated that, for a PWR plant, the probability of a small-small LOCA ($0.38 \text{ in.} < D \leq 1.2 \text{ in.}$) due to bolting failure would be $4.39\text{E-}3/\text{reactor year}$. If the PNL estimate was correct, during almost 6 years of operation of 80 PWR plants since the issuance of the PNL value-impact analysis, one would expect the occurrence of $6 \times 80 \times 4.39\text{E-}3$, or about two small-small LOCA due to failure of bolting on the RCPB. Actually, no LOCAs of any size have been experienced during this period. This tends to confirm the conclusion discussed

earlier in Section 5.1 that the PNL risk estimation was conservatively high.

5.2.3 Survey of Reports on Precursors to Potential Severe Core Damage Accidents

NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: A Status Report" (Refs. 5-9), is an annual report that documents the findings of the Accident Sequence Precursor (ASP) Program conducted by the Nuclear Operations Analysis Center at Oak Ridge National Laboratory (ORNL). The ASP Program reviews the Licensee Event Reports (LERs) of operational events that have occurred at nuclear power plants and identifies and categorizes precursors to potential severe core damage accident sequences. Accident sequences of interest are those that, if additional failures had occurred, would have resulted in inadequate core cooling with the potential for severe core damage. Accident sequence precursors are events that are important elements in such accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition leading to severe core damage.

A search was performed by ORNL to list all LERs during the period from 1985 to 1989 that mentioned bolting or threaded fasteners. These LERs were compared to the list of precursors identified in NUREG/CR-4674 as having a conditional core damage probability greater than $1E-6$ during the same time period. Only the 14 LERs summarized in Table 1 were identified. Among the 14 cases listed, bolting problems are generally only partial contributors to the precursors. A large number of these bolting problems were related to internals bolting, others are related to improper torquing, loose bolts and nuts, missing bolts, etc. The recommended bolting integrity program delineated in the resolution of GSI-29 is not designed to address all of these types of bolting problems. Problems related to loose or improperly aligned internal set screws, missing bolts, loose bolts or nuts, etc., are better handled by such programs as inservice inspection and regular maintenance or the implementation of other on-going NRC programs such as Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants," and Individual Plant Examinations for External Events (IPEEE) that address the inadequacies of supports and their bolting due to design and installation.

5.3 Other Considerations

When the scope of GSI-29 was limited to the RCPB, another GSI was established to cover bolting associated with other components, particularly structural supports where SCC had led to failures. This issue, GSI-62, "Reactor Systems Bolting Application," was

reevaluated by the NRC staff in August, 1988. It was concluded that the safety concerns of GSI-62 would be addressed under the broadened scope of GSI-29. Therefore, GSI-62 is considered subsumed by the resolution of GSI-29.

6. IMPLEMENTATION

This regulatory analysis provides bases for the resolution of GSI-29 and recommends (1) issuing a generic letter for information to plants having an OL or CP and (2) developing a new SRP section dealing with safety-related threaded fasteners for future plants.

It is expected that recipients of the generic letter will review the information cited in the generic letter for applicability to their facilities and consider appropriate actions, if necessary, to avoid future problems. However, the suggestions transmitted by the generic letter do not constitute NRC requirements; therefore, no specific action or written response is required by the generic letter.

For future plants, a more effective review in a future SRP section should be used. The elements of the review would include the design of all safety-related joints, the selection of threaded fastener materials, and programmatic aspects dealing with bolting integrity during construction and operation/maintenance.

7. **REFERENCES**

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3. EPRI NP-5067, "Good Bolting Practices--A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual," and Volume 2: "Small Bolts and Threaded Fasteners," EPRI, 1987 and 1990, respectively.
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7. NUREG/CR-4674, Volume 7 and Volume 8, "Precursors to Potential Severe Core Damage Accidents: 1987, A Status Report," US NRC, July 1989.
8. NUREG/CR-4674, Volume 9 and Volume 10, "Precursors to Potential Severe Core Damage Accidents: 1988, A Status Report," US NRC, February 1990.
9. NUREG/CR-4674, Volume 11 and Volume 12, "Precursors to Potential Severe Core Damage Accidents: 1989, A Status Report," US NRC, August 1990.

TABLE 1

LEERS ('85 - '89) RELATED TO THREADED FASTENERS
THAT RESULTED IN CONDITIONAL CD PROBABILITY $\geq 10^{-6}$

LER No.	Conditional CD probability	Description
1. 333/85 - 025	1.1×10^{-6}	A loosened locking screw in a motor operator on the HPCI steam supply isolation valve caused the torque switch to be set improperly, resulting in failure of the valve and unavailability of HPCI. Since RCIC was also unavailable due to maintenance, plant was placed in 24 hour LCO.
2. 373/85 - 045	7.2×10^{-5}	Loss of Circulating and Non-Safety Service Water, due to fatigue failure of circulating pump discharge valve gear operator mounting bolts (Installation error and wrong torque value).
3. 249/86 - 013	2.7×10^{-6}	2/3 EDG failed to close manually onto bus 33-1 due to loose terminal block screw in a junction box, but was able to be synchronized manually to bus 23-1.
4. 389/86 - 011	2.6×10^{-6}	Train "2A" EDG failed to start because of mechanical problem with the governor. Train "2B" EDG was also shutdown due to rubbing between its cooling fan and the shroud. This was caused by loosening of fan hub set screw resulted from vibration of cooling unit. Therefore EDG system was not available.
5. 293/87 - 014	3.9×10^{-4}	LOOP and 1 EDG out of service (inspection). The pre-lube pump of 2nd EDG failed due to loose mounting bolts.

LERs ('85 - '89) RELATED TO THREADED FASTENERS
THAT RESULTED IN CONDITIONAL CD PROBABILITY $\geq 10^{-6}$ (Cont'd)

LER No.	Conditional CD probability	Description
6. 324/87 - 001	2.4×10^{-4}	Reactor trip with HPCI unavailable, RCIC full flow test isolation valve failed to close (50% open) due to out of adjustment of limit switch, caused by improperly aligned set screw.
7. 280/88 - 011	1.5×10^{-5}	PORV failure due to incorrectly torqued hold-down screws and bolts which allowed actuator diaphragm to shift.
8. 321/88 - 018	1.5×10^{-5}	Reactor scram with loss of nonessential loads and RCIC degraded operation (RCIC turbine steam supply valve failed to close fully caused by unsecured set screw for yoke stem bushing).
9. 323/88 - 008	4.1×10^{-5}	LOOP with safety injection. Galled thread resulted in RC pump feeder line electrical ground fault.
10. 328/88 - 005	3.8×10^{-4}	Both train A and B centrifugal charging pump speed increasers failed due to back out of gland seal retaining bolts inside the lube oil pump, resulting in inavailability of high head injection system.
11. 339/88 - 004	2.5×10^{-5}	Both EDGs unavailable, one out for maintenance, other the output breaker failed to close (because closing springs were not charged, due to mounting bolts on the charging motor backing out of breaker housing, allowing the charging motor to disengage).

LERs ('85 - '89) RELATED TO THREADED FASTENERS
THAT RESULTED IN CONDITIONAL CD PROBABILITY $\geq 10^{-6}$ (Cont' d)

LER No.	Conditional CD probability	Description
12. 324/89 - 009	3.6×10^{-5}	Reactor scram caused a LOOP. One of two trains of LPCI/RHR was inoperable due to stuck-closed injection valve, caused by unscrewing of valve disc nut (inadequate insertion of locking pin).
13. 400/89 - 006	4.4×10^{-6}	Several mounting bolts connecting the flanged junction box to the "B" main feedwater pump (MFP) were missing. Incorrect action by Fire Protection Technicians subsequent to difficulty in resetting fire protection deluge valves over the MFPs resulted in water sprayed on the "B" MFP, causing internal short in the junction box which led to reactor trip/turbine trip.
14. 483/89 - 008	1.2×10^{-6}	During a turbine trip, the failure to reset a protective relay (due to a loose calibration set screw) for the main generator output breaker led to complete loss of power to the 4KV Safeguards Bus. This resulted in inoperable radiation monitors and also ESF actuation of turbine driven aux feed pump.

Total
(5 yrs. 100 reactors) 1.23×10^{-3}

Average Conditional CD probability = $1.23 \times 10^{-3}/500 = 2.46 \times 10^{-6}/RY$

APPENDIX A

BOLTING DEGRADATION
VALUE-IMPACT ANALYSIS

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BOLTING DEGRADATION VALUE-IMPACT ANALYSIS

1.0 INTRODUCTION

Crucial bolting applications in nuclear power plants constitute an integral part of the reactor coolant pressure boundary (RCPB) and include closure studs or bolts on reactor vessels, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and jeopardize safe operation of the plant. Other bolting applications, such as component support and embedded anchor bolts or studs, are essential for withstanding transient loads created during abnormal or accidental conditions.

In recent years, the number of bolting degradation events (bolt cracking, corrosion, failure, etc.) reported by licensees of operating reactors has increased (NRC 1983a, NRC 1985). Many of these events are related to RCPB applications and major component support structures. This has caused an increasing concern regarding the integrity of the RCPB and the reliability of the component support structures following a loss-of-coolant accident (LOCA) or seismic event.

Most of these bolting degradations were discovered either during refueling outages or scheduled inservice inspections or maintenance/repair outages. Therefore, these reported incidents have had no immediate impact on public health and safety. No accidents have yet resulted from these bolting degradations. However, degradation or failure of these studs or bolts could constitute a reduction in the integrity of the RCPB. Concern is compounded by the fact that there is currently no reliable ultrasonic testing (UT) method to detect the cracking or degradation which are principally caused by stress corrosion cracking (SCC) and borated-water corrosion (BWC). Visual examination is currently the only viable method to discover degradation by BWC and surface inspection is currently the best inspection method to detect stress corrosion cracking. A LOCA could conceivably occur due to undetected bolting failure within the RCPB.

2.0 PROPOSED ACTION AND POTENTIAL ALTERNATIVES

This section presents the proposed action considered in the value impact analysis, including a discussion of potential alternatives.

2.1 PROPOSED ACTION

To minimize potential bolting problems in nuclear power plants, improvements in one or all of the following areas could be recommended: bolt design, materials used, fabrication method, installation and inservice inspection.

Because of concern about increasing bolt degradation, the NRC is considering the need for providing specific guidance to the industry in the form of a section of the Standard Review Plan (SRP) on RCPB bolting. The purpose of this is to prevent the two most common failure modes of bolts. These are stress corrosion cracking (SCC), which was identified as the most common mode of bolt failure in past data (NRC 1983a), and boroated water corrosion (BWC), which was the dominant failure mode of bolts acquired from recent data (NRC 1985).

The proposed regulatory revised approach for reducing SCC in bolts is to use a material with the lowest strength consistent with the required clamping force of the jointed connection, and to apply a high accurate preload. Incorporating these materials into their proper joint design could involve improving control of bolt fabrication, source and receiving inspection, and installation. Discussion and applications for improvement in these areas can be found in NUREG/CR-3604 (NRC 1984).

In bolting applications subject to BWC, the proposed SRP suggests use of an alloy or specialized coatings resistant to BWC (e.g., stainless steel), or use of the "corrosion allowance" concept. This concept basically involves using a bolt larger than the actual design requirements, in order to allow for corrosion that may take place before the bolt is next inspected. For example, if a jointed connection design calls for 2 in. diameter bolts, then 2½ in. diameter bolts would be used. This allows for a corrosion reduction of the bolt diameter to 2 in. (e.g., based on average leak and corrosion rates) during the time period before the connection is inspected and serviced. Thus, the bolt size should not fall below the original design requirement.

Discussion with NRC personnel on the proposed action pointed out that implementing a "corrosion allowance" design change to a backfit plant would be very impractical. This would involve a tremendous amount of backfitting (e.g., redesign, re-tapping of bolted connections, etc.) and would not be a practical method for dealing with bolting degradation in operating plants.

Thus, the proposed action for this value impact analysis (VIA) is to replace bolts or studs in RCPB bolting applications with stainless steel (SS) materials that are more resistant to corrosion than the materials presently used. Inservice inspection requirements for RCPB bolting would remain as described in ASME Code XI (ASME 1983). The basic benefit of this proposed action would be to decrease the probability of having corroded bolts or studs, which would increase the integrity of the RCPB. Costs would include the cost for SS bolting materials and incremental labor for bolt replacement.

The proposed SRP applies to RCPB bolting only (excluding the reactor pressure vessel) as does this VIA. RCPB bolting includes jointed connections such as in reactor coolant pump case and seal flanges, valve bonnets, and steam generator manway covers in pressurized water reactors (PWRs). Bolting for piping, equipment, or component supports is not considered. The proposed action involves replacing all RCPB non-stainless steel bolts or studs with stainless steel, or with a material more resistant to corrosion than stainless steel in specific applications. Materials for these special applications could include 17-4 PH or various types of Inconel.

2.2 PROPOSED ALTERNATIVES

In NUREG-0933, the NRC evaluated ways to improve the efficiency and adequacy of inservice inspection (NRC 1983b). They stated, "...that the proposed fix could vary considerably depending on the type and depth of solution sought. However, the most probable fix is to visually inspect bolts or studs using an improved ultrasonic (UT) inspection technique and a more frequent inspection schedule".

Using improved inspection techniques and more frequent inspections may be a viable approach. This would encompass developing the inspection technology to an established degree of reliability and incorporating changes in the inservice inspection requirements affecting RCPB bolting applications. The basic benefit of this could be a higher probability of detecting SCC and BWC in bolts or studs before extensive failure occurred.

Current inservice inspection requirements for Class 1 (RCPB) applications call for a volumetric examination (UT)^(a) only for all bolts and studs over 2 inches diameter (except for reactor closure studs). Surface examination (MT or PT)^(a) is now required only for reactor vessel studs, and only if removed. For Class 1 bolts, studs, and nuts that are 2 inches nominal diameter or less, only a visual examination (VT-1) is required. Irrespective of nominal size for all Class 1 bolts, studs, nuts, bushings, and washers, the specified inspection is required during each inspection interval.

Except for reactor coolant pump stud corrosion, most failures have occurred in bolts or studs 2 inches diameter or less. Present Code UT procedures are not sufficiently sensitive to detect initiation of SCC, and the use of special UT techniques and calibration standards based on notch reflectors to simulate critical flaw parameters and enhance detection reliability are required. Furthermore, existing field technology offers no reliable means to ultrasonically detect BWC of bolts, except for large diameter bolts that have heater holes.

At the present time, visual examination to detect BWC may require insulation removal and/or disassembly of the component.

(a) UT = Ultrasonic Examination
 MT = Magnetic Particle Examination
 PT = Liquid Penetrant Examination

In summary, the current practices and requirements for inservice inspection of RCPB bolting (excluding the reactor vessel) are not particularly effective in detecting SCC and corrosion wastage. UT required for bolts with a diameter greater than 2 in. does not reliably detect SCC or BWC. Visual examination of bolting with a 2 in. diameter or less can detect BWC if good surveillance techniques are employed, but the efficiency of this technique is questionable for detecting SCC initiation. The potential alternative of using an improved UT technique would only affect large diameter bolting unless code requirements are changed. Also, reliable UT techniques may take years to develop. Using a more frequent inspection schedule may detect more bolts affected by BWC before they fail but will not necessarily improve detection of SCC. Therefore, bolt replacement was considered the viable alternative in this VIA.

A factor considered in estimating risk reduction and costs resulting from this alternative is the number of times this replacement of bolts will occur. The proposed action is to replace all bolts once in the RCPB with SS materials during one interval of the inservice inspection schedule. The ASME Code currently calls for two inspection programs, each entailing four inspection intervals over a 40 year plant life. Based on a more accepted plant life of 30 years, 3 inspection intervals would result. Thus, pertinent RCPB bolting could be inspected and serviced a minimum of 3 times over the life of a plant. Considering only backfit plants, some have already gone through the first inspection interval. The exact number of plants that have gone through inspection would depend upon which inspection program is being followed and on the decreases and/or extensions of inspection periods encountered by the plant due to coinciding inspections with plant outages. The first interval ends after 3 calendar years of plant service in Program A and after 10 calendar years in program B. Considering that about one-half of the 47 PWRs affected by this action started commercial operation > 10 years ago, some plants could have 2 inspection intervals left and some 3. This analysis assumes that all RCPB bolting will be replaced once with SS materials over a ten year interval. Subsequent bolt replacement (if any) would be according to the inservice inspection requirements of the ASME Code and is not considered in this VIA.

3.0 AFFECTED DECISION FACTORS

Parameters in a VIA and those affected in this study are shown in Table 3.1

TABLE 3.1. Checklist for Identification of Affected Attributes

<u>Decision Factors</u>	<u>Causes Quantified Change</u>	<u>Causes Unquantified^(a) Change</u>	<u>No Change</u>
Public Health	X		
Occupational Exposure (Accidental)	X		
Occupational Exposure (Operational)	X		
Offsite Property	X		
Onsite Property	X		
Regulatory Efficiency		X	
Improvements in Knowledge		X	
Industry Implementation	X		
Industry Operation	X		
NRC Development	X		
NRC Implementation	X		
NRC Operation			X

(a) Not readily estimated in dollars.

4.0 VALUE-IMPACT ASSESSMENT SUMMARY

This section contains a summary of the best, high, and low estimates for the value and impact decision factors affected in this study. Results are shown below in Table 4.1.

TABLE 4.1. Value-Impact Assessment Summary - Total for All Plants

<u>Decision Factors</u>	<u>Best Estimate</u>	<u>High Estimate</u>	<u>Low Estimate</u>
<u>Values^(a)</u> (man-rem)			
Public Health	10,063	25,988	0
Occupational Exposure (Accidental)	76.2	337.7	0
Occupational Exposure (Operational)	-320	-1290	-60
Regulatory Efficiency	N/A		
Improvement in Knowledge	N/A		
Total Quantified Value	9819	25,036	-60
<u>Impacts^(b)</u>			
Industry Implementation ^(d) Cost	4.7E+05	1.5E+06	1.8E+05
Industry Operating Cost	1.8E+06	2.6E+06	1.5E+06
NRC Development Cost ^{(c)(d)}	5.0E+04	1.5E+05	1.7E+04
NRC Implementation Cost ^(d)	2.9E+04	8.7E+04	9.7E+03
NRC Operation Cost	0	0	0
Public Property	-2.2E+06	-3.1E+07	0
Onsite Property	-1.3E+06	-5.4E+06	0
Total Quantified Impact	-1.2E+06	-3.2E+07	1.7E+06

(a) A decision term is a value if it supports NRC goals. Principal among these is the regulation of safety.

Note: The footnotes for this table are continued on the next page.

- (b) Impacts are defined as the costs incurred as a result of the proposed action. Negative impacts indicate cost savings. A 10% discount rate is used.
 - (c) Does not include costs expended to date.
 - (d) Does not include any discounting calculations (cost).
- N/A = Not affected.

5.0 UNQUANTIFIED RESIDUAL ASSESSMENT

There are no unquantified attributes in the assessment of this action.

6.0 DEVELOPMENT OF QUALIFICATION

6.1 PUBLIC HEALTH

A risk analysis was performed to assess the effects of replacing bolts or studs in the RCPB of backfit plants with SS materials more resistant to corrosion than the materials currently being used. This was accomplished by examining accident sequences in representative PWR and BWR plants that involve LOCA initiating events. If bolting failures in a RCPB jointed connection did cause a LOCA, the resulting LOCA would probably be quite small, analogous to a reactor coolant pump seal failure as designated in the Arkansas Nuclear One (ANO-1) IREP study (Kolb et al, 1982). Probabilities for larger LOCAs due to bolting failure are assumed to be smaller.

Water hammer (with or without coincident bolting failure) is not considered a potential LOCA initiator in the subsequent analysis because the only water hammer events in the primary system have occurred in the pressurizer relief discharge line. Thus, no increase in LOCA frequency from water-hammer-induced bolting failures is assumed. Water-hammer-induced bolting failures may affect the availability of other systems which can have an effect upon core-melt frequency. However, a review of the frequencies of water-hammer events in various systems of a PWR estimated by Amico and Ferrell (1982) indicate that the RCPB system is not involved (NRC 1982). Similarly, a review of a methodology developed for estimating public risk for seismic external events, (Gallucci, 1983) which is based upon the seismic risk analysis for the Zion PWR (Commonwealth Edison, Co., 1981), was made. The conditional failures of six items were identified as potentially affecting the seismic core-melt frequency in a significant way. None of these items involve RCPB bolting applications. Thus, only LOCA initiating events were assumed to be affected by RCPB bolting failures.

6.1.1 PWR Risk Reduction

To calculate risk reduction due to bolt replacement with SS materials resistant to corrosion, it was first necessary to determine frequencies for bolting failure events per reactor-year and for LOCAs per bolting failure event. The bolting failure event frequency was estimated from the most recent data available on bolting degradation in PWRs. This data, which will be published in NUREG-1095 (NRC 1985), was a result of licensees' responses to IE Bulletin 82-02 (NRC 1982). Data were reported on threaded fastener degradation in the PR RCPBs and included such items as the number of connections inspected, the number of connections leaking, the number of connections where bolt or stud corrosion had occurred, and the number of connections that had (one or more) bolts replaced due to corrosion. The numbers of interest used in this risk analysis were 27 corrosion-related bolt replacement events that occurred in 41 PR plants over about 1.5 years. This results in a bolting failure frequency of 0.439 per reactor year (27 bolting failure events/41 plants x 1.5 years).

It was conservatively assumed that a bolt was in a failed state when it was replaced (i.e. a bolt that did not meet the inservice inspection requirements of the ASME Code, or that was replaced for some other reason was assumed to be a bolt failure). Though other failure modes for bolt degradations, including mechanical damage, destruction by removal, and steam erosion were identified, the only bolt replacement data reported were on connections where BC occurred. A note of interest is that no SCC was found in the 414 connections inspected and that all corrosion of bolts was directly due to leakage of reactor coolant from failed gaskets and/or inappropriate torques on bolts (NRC 1985).

The observed trend is that the frequency of RCPB bolting failures in operating reactors is increasing. The bolting failure frequency calculated above (0.439/ry) is a significant increase over past data. NUREG-0943, which reported bolting degradation events from October 1964 to March 1982, cited 19 bolt failure events that involved the RCPB (NRC 1983a). This occurred over about 350 reactor years resulting in a 0.0543 bolting failure frequency per reactor year. This report also stated that the number of degraded bolt incidents had increased over the last two years (i.e. March 1980 to March 1982). This increase has continued at a growing rate. The 0.0543 bolting failure frequency per reactor year increased by eight times to 0.439 over the time period of the most recent data reported in NUREG-1095 (i.e. August 1982 to February 1984).

Both a high and best estimate for LOCA frequency per bolting failure was used in calculating PR risk reduction. The low estimate was assumed to be zero. This is attributed to the fact that no LOCAs due to bolting failures have yet occurred. The high estimate was determined using the number of corrosion-related bolt replacements events from recent data (NRC 1985), and corresponding data from past events (NRC 1983a). The 27 bolt replacement (failure) events from recent data were a combination of BC failures in steam generator manway cover, reactor coolant pump case and seal flange, valve packing, and valve body to bonnet connections. Eleven analogous BC failure events occurred out of the 44 degradation events reported in NUREG-0943. Out of these 38 (27 + 11) failure events no LOCAs have occurred. It was assumed that an estimate of 1 LOCA per 39 bolting failure events (0.026 LOCAs/bolt failure) would be a high frequency estimate.

The best estimate used in this analysis for LOCAs per bolting failure was 0.01. This was based on NUREG-1095 which states that the assumption of one small LOCA per 10 corrosion events, which was used in the risk reduction analysis of bolting failure in NUREG-0933, appears to be quite conservative, and that this rate could be lowered by a factor of ten or more to compensate for the increase in bolting failures in recent years.

The best estimate of LOCA frequency per bolting failure used in this analysis may still be conservative (i.e. high). The use of factors of safety in the design of bolted connections and the continued emphasis on improving installation, maintenance and inspection techniques all contribute to preventing bolting degradation. In addition, the use of RCPB leak detection devices help plants to detect leaking connections, so they can be fixed before developing significant corrosion and bolt failure. However, it is yet to be seen whether these measures reverse the growing trend of bolting failure frequency.

The frequency for bolting failures per reactor year was multiplied by the high and best estimates for LOCA probability per bolting failure to obtain the high and best case of LOCA frequencies. The base case parameter values were obtained by adding the additional probabilities of 0.0114 (high) and 0.00439 (best) to the original small-small LOCA probability of the ANO-1 risk assessment (0.02). The original probabilities for this analysis' affected parameters are shown in Table 6.1. These additional high (0.0114) and best (0.00439) probabilities for the B(1.2) LOCA case were then proportioned to the ratio of the B(1.2) to B(1.66) and B(1.2) to B(4) LOCA probabilities, in order to determine their additional contribution to the original B(1.66) and B(4) LOCA frequency cases. It was assumed that the proposed action would essentially eliminate bolting failures in the RCPB.

Thus, the additional LOCA probabilities (due to bolting failure), used in determining the base case parameters' probabilities would also be eliminated. Therefore, the adjusted case probabilities of the affected parameters would be the same as the original probabilities shown in Table 6.1. Table 6.2 lists the frequencies of the affected accident sequence values from the original ANO-1 study, the base case, and adjusted case for the high and best estimates.

The contributions of these sequences to individual release categories were then summed to obtain the frequency for each PWR release category. Results of this calculation are shown in Table 6.3 for the High and Best estimates for the B(1.2) case.

The final step in the risk calculation is to multiply each of the changes in release category frequency from the base to adjusted case by the dose factor for each release category. The dose factors used in this analysis assume population density of 340 people per square mile at a midwestern site (Andrews et al. 1983). The factors are shown in Table 6.4.

TABLE 6.1. Potential PWR Bolt Degradation Affected Parameters

Parameter	Original & Adjusted Case Probability/Frequency	Base Case Probability/Frequency	
		Best	High
B(1.2) - reactor coolant pump seal failure or small-small LOCA	2.0E-02/PY	2.4E-02/PY	3.1E-02/PY
B(1.66) - reactor coolant system pipe break 1.2" < D ≤ 1.66"	3.1E-04/PY	3.8E-04/PY	4.9E-04/PY
B(4) - reactor coolant system pipe break 1.66" < D ≤ 4"	3.8E-04/PY	4.6E-04/PY	6.0E-04/PY

The risk reduction results for the three LOCA cases are shown in Table 6.5 for the high and best estimates. The high estimates were taken as the sum of the high estimate LOCA contributions. Best estimates were taken as the sum of the best estimate LOCA contributions. Through engineering judgment, the low estimates were determined to be zero. Results are shown in Table 6.6.

6.1.2 BWR Risk Reduction

Grand Gulf 1 was assumed to be the representative plant in BWR calculations for risk and core-melt frequency reduction. Results of the Grand Gulf 1 RSSMAP study used for prioritization of safety issues (Andrews et al. 1983) were assumed to be representative of all backfit BWRs. As with the PWR plants, bolting failures in the RCPB of BWRs are assumed to affect (i.e., increase the probability of) LOCA initiating events. Thus, the affected parameter for potential BWR bolt degradations is "S". This parameter corresponds to a small LOCA (the break area is less than one square foot) and has a frequency of 0.0014/py in the Grand Gulf 1 risk assessment. The affected dominant accident sequence is then "SI", which is associated with the BWR1 and BWR2 release categories (Andrews et al, 1983).

As in the PWR risk reduction analysis, the proposed action is assumed to essentially eliminate the problem of bolting failure due to corrosion in RCPB applications. Thus, the increased base case frequency for core-melt, due to the additional probability of bolting failures to induce LOCAs, is reduced to the adjusted case by implementing the proposed action. The additional frequency of LOCAs per reactor year was calculated as follows.

TABLE 6.2. Affected ANO-1 Accident Sequence Frequencies -
High and Best LOCA/PY Estimate Cases

Affected Accident Sequence	Accident Frequency (events/reactor year)		
	Original	Base	Adjusted
High Case			
B(1.2)D	2.8E-06	4.4E-06	2.8E-06
B(1.2)DC	4.4E-06	6.9E-06	4.4E-06
B(1.66)H	1.2E-06	1.9E-06	1.2E-06
B(4)H	1.4E-06	2.2E-06	1.4E-06
Best Case			
B(1.2)D	2.8E-06	3.4E-06	2.8E-06
B(1.2)DC	4.4E-06	5.4E-06	4.4E-06
B(1.66)H	1.2E-06	1.5E-06	1.2E-06
B(4)H	1.4E-06	1.7E-06	1.4E-06

TABLE 6.3. Affected PWR Release Category Frequencies for B(1.2) Case
Release Category Frequencies (event/reactor-yr)

<u>Release Category</u>	<u>Original</u>	<u>Base</u>	<u>Adjusted</u>
High Estimate			
PWR1	2.0E-08	1.3E-09	8.1E-10
PWR2	2.0E-05	7.2E-06	4.6E-06
PWR3	1.0E-07	0	0
PWR4	2.0E-07	6.7E-08	4.3E-08
PWR5	2.0E-07	4.1E-08	2.6E-08
PWR6	2.0E-05	5.7E-06	3.6E-06
PWR7	1.0E-05	2.1E-06	1.3E-06
Best Estimate			
PWR1	2.0E-08	9.9E- 10	8.1E-10
PWR2	2.0E-05	5.6E-06	4.6E-06
PWR3	1.0E-07	0	0
PWR4	2.0E-07	5.2E-08	4.3E-08
PWR5	2.0E-07	3.2E-08	2.6E-08
PWR6	2.0E-05	4.4E-06	3.6E-06
PWR7	1.0E-05	1.6E-06	1.3E-06

Through engineering judgment, the low estimate was determined to be zero. This assumes that no increase in small LOCA initiating events will be due to bolting degradation in the RCPB. This is attributed to the fact that no LOCAs have occurred in operating BWRs due to the result of bolt failure. In addition, BWRs do not have significant BWC problems as do PWRs because of the differences in reactor coolant chemistry.

The best estimate was partially derived from data in a report on threaded fastener experience in nuclear power plants (NRC 1983a). Only three bolting degradation events over 350 reactor years occurred in BWRs out of the 44 events covered in this report. The remaining 41 events were in PWRs. These three BWR events involved reactor vessel closure studs, reactor vessel thermal

TABLE 6.4. PWR Dose Conversion Factors

<u>Release Category</u>	<u>Dose Factor (man-rem)</u>
PWR1	5.4E+06
PWR2	4.8E+06
PWR3	5.4E+06
PWR4	2.7E+06
PWR5	1.0E+06
PWR6	1.5E+05
PWR7	2.3E+03

TABLE 6.5. PWR Risk Results for All LOCA Cases

<u>LOCA Case</u>	<u>Core-Melt Frequency Reduction (events/py)</u>	<u>Public Risk Reduction (man-rem/py)</u>
High Estimate		
B(1.2)	5.5E-06	12.9
B(1.66)	7.8E-07	2.1
B(4)	9.1E-07	2.5
Best Estimate		
B(1.2)	2.1E-06	5.0
B(1.66)	3.0E-07	0.83
B(4)	3.3E-07	0.92

TABLE 6.6. PWR Risk Results

<u>LOCA/PY Case</u>	<u>Core-Melt Frequency Reduction (events/py)</u>	<u>Public Risk Reduction (man-rem/py)</u>
High Estimate	7.2E-06	17.5
Best Estimate	2.7E-06	6.8

shield bolts, and motor hold down bolts on valve limit-torque operators. This bolting failure frequency of $8.6\text{E-}03$ per reactor year (3 bolt degradation events/350 py) was multiplied by the best estimate for LOCAs per bolting failure used in the PWR risk reduction analysis (0.01). This results in an additional probability of $8.57\text{E-}5$ LOCAs/PY due to bolting failures.

The High estimate used the same bolting failure per reactor year frequency of $8.6\text{E-}3$, but was multiplied by the High estimate for LOCAs per bolting failure used in the PWR risk reduction analysis (0.026). This results in an additional probability of $2.23\text{E-}4$ LOCAs/PY for the High estimate case. Table 6.7 indicates the High and Best estimates for the original, base, and adjusted case frequencies for the affected accident sequence.

The accident frequency was then multiplied by its containment failure mode probabilities to obtain frequencies for the BWR release categories. Results of these calculations are shown in Table 6.8.

The difference between the base and adjusted-case frequencies was then multiplied by the dose conversion factor for each release category. These products were then summed to yield the public risk reduction. The dose conversion factors are shown in Table 6.9, and the risk reduction results are shown in Table 6.10.

TABLE 6.7. Affected Grand Gulf 1 Accident Sequence Frequencies

<u>Affected Accident Sequence</u>	<u>Accident Frequency (events/reactor-yr)</u>		
	<u>Original</u>	<u>Base</u>	<u>Adjusted</u>
High			
SI	4.6E-06	5.3E- 06	4.6E-06
Best			
SI	4.6E-06	4.9E-06	4.6E-06

TABLE 6.8. Affected Release Category Frequency (Events/Reactor-yr)

<u>Release Category</u>	<u>Original</u>	<u>Base</u>	<u>Adjusted</u>
High			
BWR1	1.1E-07	5.9E-08	5.1E-08
BWR2	3.4E-05	5.4E-06	4.7E-06
Best			
BWR1	1.1E-07	5.4E-08	5.1E-08
BWR2	3.4E-05	5.0E-06	4.7E-06

TABLE 6.9. BWR Dose Conversion Factors

<u>Release Category</u>	<u>Whole Body Dose Factor (man-rem)</u>
BWR1	5.4E+06
BWR2	7.1E+06

TABLE 6.10. BWR Risk Results

<u>LOCA/PY Estimate</u>	<u>Core-Melt Frequency Reduction (per PY)</u>	<u>Public Risk Reduction (man-rem/reactor-yr)</u>
Best Estimate	2.9E-07	2.0
High Estimate	7.5E-07	5.3
Low Estimate	0	0

6.1.3 Summary of Risk Reduction

To estimate the public safety benefits for this proposed action, the results for a single plant need to be summed over the remaining lives of the affected plants. Results of the ANO-1 and Grand Gulf 1 analyses along with plant characteristics from Table 6.11 were used. The results are shown in Table 6.12.

TABLE 6.11. Plant Characteristics Assumed for Evaluation Purposes in this Analysis

<u>Consideration</u>	<u>Affected PWR</u>	<u>Backfit BWR</u>	<u>Plants TOTAL</u>
Number of Plants	47	24	71
Remaining lives in years	27.7	25.2	26.9

TABLE 6.12. Risk Results - All Plants

	<u>Public Risk Reduction All Affected Plants (man-rem)</u>	<u>Core-Melt Frequency Reduction-All Affected Plants (events/py)</u>
PWR		
Best Estimate	8853	2.7E-06
High Estimate	22,783	7.2E-06
Low Estimate	0	0

Note: Table 6.12 is continued on next page.

Table 6.12. (Con't)

BWR		
Best Estimate	1210	2.9E-07
High Estimate	3205	7.5E-07
Low Estimate	0	0
TOTAL		
Best Estimate	10,063	1.9E-06 (a)
High Estimate	25,988	5.2E-06 (a)
Low Estimate	0	0

(a) These results are the weighted average, according to the number of plants and remaining lives, between the corresponding PWR and BWR estimates. An example calculation is as follows:

For the best estimate case,

$$\begin{aligned}
 2.7\text{E-}06/\text{py} \times 47 \text{ p} \times 27.7 \text{ yr} &= 3.5\text{E-}03 \\
 &+ \\
 2.9\text{E-}07/\text{py} \times 24 \text{ p} \times 25.2 \text{ yr} &= \underline{1.8\text{E-}04} \\
 &3.7\text{E-}03
 \end{aligned}$$

$$3.7\text{E-}03/[(47 \text{ p} \times 27.7 \text{ yr}) + (24 \text{ p} \times 25.2 \text{ yr})] = 1.94\text{E-}06/\text{p-yr}$$

6.2 OCCUPATIONAL EXPOSURE - ACCIDENTAL

Occupational exposure from accidents can be estimated as the product of the change in total core-melt frequency and the occupational exposure likely to occur in the case of a major accident. The change in core melt frequency has been estimated to 1.9E-06/PY and 5.2E-06/PY for the best and high estimates, respectively.

The occupational exposure in the event of a major accident has two components. The first is the "immediate" exposure to the personnel onsite during the span of the event and its short term control. The second part is the longer term exposure involved with the cleanup and recovery from the accident.

The total avoided occupational exposure is calculated as follows:

$$D_{\text{TOA}} = \text{NTD}_{\text{OA}}; D_{\text{OA}} = \Delta F(D_{\text{IO}} + D_{\text{LTO}})$$

where:

D_{TOA} = total avoided occupational dose

N = number of affected facilities

T = average remaining lifetime

D_{OA} = avoided occupational dose per reactor year

ΔF = change in core-melt probability

D_{IO} = "immediate" occupational dose

D_{LTO} = long-term occupational dose

Results of the calculations are shown in Table 6.13. Uncertainties are conservatively propagated by use of extremes (e.g., high estimate D_{IO} + high estimate D_{LTO}).

TABLE 6.13. Accidental Occupational Dose

	Reduction in Core Melt Frequency (events/py)	Immediate ^(a) Occupational Dose (man-rem/event)	Long Term ^(a) Occupational Dose (man-rem)	Total Avoided Occupational Exposure (man-rem)
Best Estimate	1.9E-06	1E+03	2E+04	76.2
High Estimate	5.2E-06	4E+03	3E+04	337.7
Low Estimate	0.0		1E+04	0.0

(a) Based on cleanup and decommissioning estimates, NUREG/CR-2601 (Murphy and Holter, 1982)

6.3 OCCUPATIONAL EXPOSURE - OPERATIONAL

Occupational exposure will result from performing inservice inspection and maintenance on RCPB bolted connections. Exposure rates can be expected to be higher than for performing work on other bolted connections since the majority of the connections will be inside of containment and have been in close contact with reactor coolant. This proposed action can affect occupational exposure. This effect will result from bolt replacement and not from upgrading receiving inspection for bolt materials. The amount of this effect will depend upon the conditions of how the bolts or studs of the connection are to be examined during inservice inspection. For example, Examination Category B-G-2, "Pressure Retaining Bolting, 2 in. and less in diameter," under Table IWB-2500-1 of the ASME Code Section XI, states that bolting may be examined, 1) in place under tension; 2) when the connection is disassembled; 3) when bolting is removed (ASME 1983). If the bolts of a RCPB connection are to be examined when the bolting is removed, then the proposed action of replacing

the bolts with new bolts would not increase maintenance time above that required to bring new bolts to the work area and to discard the old bolts. This time increase is assumed to be negligible for occupational exposure and minimal for industry operation costs. However, if the bolts of a connection are to be examined in place or under tension, then replacing all the bolts with new ones would result in an increase of maintenance time, and thus, occupational exposure. The three conditions of bolt examination used would depend upon a plant's specific inspection schedule.

Other variables to be considered in determining occupational exposure from this proposed action involve the number of pertinent RCPB connections, the dose rates associated with working on these connections, and the increase in the amount of time and number of laborers needed to replace bolts on these connections. All of these parameters may (and probably do) vary from plant to plant.

The exact number of RCPB bolted connections in the 71 backfit plants (47 PWRs, 24 BWRs) affected by the proposed action is not known. Typical bolted connections of concern include steam generator manway covers (PWRs only), pump case and seal flanges, and body to bonnet connections of various types of valves (e.g., isolation, safety injection). The number of these bolted connections and bolts per connection was estimated for a generic PWR and BWR.

6.3.1 PWR Occupational Exposure - Operational

It was assumed that each PWR pressurizer had one manway with sixteen - $1\frac{1}{2}$ x 5 in. studs or bolts per manway cover connection. It was assumed that each plant had an average of 3 steam generators with 2 manways per steam generator. Depending upon the model of steam generator used, the number of bolts or studs per manway cover connection appears to vary from 16 to 24. An average of twenty - 2 x 5 in. bolts or studs per manway cover was assumed for this analysis.

The number of reactor coolant pumps per PWR was assumed to be 4. Each reactor coolant pump has one casing and seal flange connection with an average of twenty - $3\frac{1}{2}$ x 5 in. and 1 x 5 in. bolts or studs per connection, respectively.

To help estimate the number of valve body to bonnet bolted connections, piping and instrument diagrams (P&IDs) of the Trojan Nuclear Plant were used (Portland General Electric 1981). Pipelines of the RCPB system were traced (inside containment only) and the number, size, and type of valves were recorded. This was done for only 3 of the 4 system loops where applicable, to correspond with the assumption of 3 steam generators per plant. The results are shown in Table 6.14. No valves in piping of 1 in. diameter or less were accounted for in risk or cost estimates, in accordance with IWB-1220 Components Exempt from Examination, of the ASME Code Section XI. This is basically because reactor coolant leakage from these components are within the capacity of makeup systems.

Given these valves, it was not known how many had bolted body to bonnet connections, and how many had other types of body to bonnet connections (e.g., screwed or sealed bonnets). This analysis assumes that all pertinent RCPB valves have bolted body to bonnet connections. Bolted connection data are shown in Table 6.15.

TABLE 6.14. Reactor Coolant System Valve Description^(a) - (Amount and Type)

Valve Type	Nominal Valve Sizes (in inches)								
	<u>1½</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>6</u>	<u>8</u>	<u>10</u>	<u>12</u>	<u>14</u>
Check	3	17	7	1	5	4	3	-	-
Gate	-	-	2	2	2	-	3	-	2
Globe	3	19	5	2	2	-	-	-	-
Safety	-	-	2	-	3	-	-	-	-
TOTAL	6	36	16	5	12	4	6	0	2

(a) Includes valves inside containment to/from the safety injection, residual heat removal and chemical volume and control charging systems.

TABLE 6.15. Bolted Connection Data for a PWR

Components	Number of Connections	Number of Bolts or Studs per Connection	Size of Bolt or Stud (inches) (Diameter x Length)
Pressurizer Manway Cover	1	16	1½ x 5
Steam Generator Manway Covers	6	20	2 x 5
Reactor Coolant Pump Case Flange	4	20	3½ x 5
Seal Flange	4	20	1 x 5
Valve Body to Bonnet Connections (Valve Size - inches)			
1½	6	4	½ x 2½
2-3	52	4	5/8 x 3
4	5	8	5/8 x 3½
6-8	16	8	3/4 x 4
10-12	6	12	7/8 x 4½
14	2	12	1 x 5

The number and size of bonnet bolts per valve can vary greatly. No data on the number and size of bonnet bolts for particular valves was readily available from valve vendors. It was assumed that the number and size of valve bonnet bolts were equivalent to the quantity and size of stud bolts for ASA steel flanged joints for corresponding pipe sizes. For example, the number and size of bonnet bolts for a 4 inch valve were assumed to be the same as for a flanged connection on a 4 inch pipe (i.e. 8 - $5/8$ x $3\frac{1}{2}$ inch bolts). This data was acquired over the range of valve sizes for a flange weighing 150 lb with a raised or male flange facing (Crane Company 1960).

Dose rates received while replacing bolts on RCPB jointed connections are assumed to occur during shutdown conditions. The dose rates used in calculating occupational exposure are shown in Table 6.16.

In determining the increase in time for bolt replacement due to this proposed action, one should note the differences in the amount of time and laborers needed to perform similar tasks. For example, a study on bolting problems reported that the manpower to position and tighten a steam generator manway cover varied from 3 men for 3 hr in one plant to 6 men for 10 hr in a second plant for the same job; a third plant which was seeking authorization to purchase new tooling, would be able to do the job with 2 men in less than 30 minutes (Loomam et al, 1981). This exemplifies the significant difference that can occur when different tools, procedures, and maintenance personnel are used. In addition, component accessibility to perform inservice inspection and maintenance (as well as unforeseen problems) significantly affect manpower requirements. All of these factors will affect the actual occupational exposure received at each plant while performing bolt replacement tasks.

This analysis assumes that about one half of the bolt replacements performed in accordance with this proposed action will occur when the connections are disassembled or when the bolting is removed. It seems very practical to replace bolts (or fulfill inservice inspection requirements for bolting) when performing maintenance on the connection components (e.g., valve bodies, pump casings, manway cover gaskets). This means that bolt replacement on about one half of the connections in Table 6.15 will contribute to occupational exposure. Bolt replacement on the other half will not affect occupational exposure of this report's proposed action, and is assumed to be done while performing other maintenance or inspection work. Note that the inservice

TABLE 6.16. Average Dose Rates Associated with
PWR Component Bolt Replacement

<u>Component</u>	<u>Dose Rate (mrem/hr)</u>
Pressurizer Manway Cover	50
Steam Generator Manway Cover	150
Reactor Coolant Pump	
Case Flange	100
Seal Flange	100
Valves	2.5

inspection code does not require that all pressure retaining bolting greater than 2 in. diameter be examined. For heat exchangers, piping, pumps, and valves, examinations are limited to bolts and studs on components selected for examination under Examination Categories for these components. This means for example, that examinations are limited to one valve within each group of valves that are of the same structural design, such as globe, gate, or check valve, and manufacturing method and that perform similar functions in the system, such as containment isolation and system overpressure protection. Thus, only bolts on one representative valve for a group of valves need be examined to meet Code requirements. What this means is that for bolting in pumps, pipes, and valves, greater than 2 in. diameter, the proposed action of bolt replacement may entail the whole replacement process (i.e., connection disassembly, bolt or stud removal, bolt replacement, reassembly, and quality control requirements). According to the bolt size estimates in Table 6.15, the only affected bolts would be on the reactor coolant pumps for PWRs. However, it is assumed that all 4 reactor coolant pump internals per plant and all the steam generator and pressurizer manway cover internals per plant will be serviced or inspected during the ten year inspection schedule. Thus, no additional time required beyond bringing new studs to the work area and discarding old studs would be warranted by this proposed action. This is because the studs would have to be removed and replaced anyway in order to perform maintenance on component internals.

This analysis assumes that the time spent in radiation zones due to the proposed action is for removing and replacing bolts or studs in about two thirds of the valve body to bonnet connections shown in Table 6.15. Thus, for the six valve component groups given in Table 6.15 (i.e. 1½ inch thru 14 inches) the following number of connections are used for calculating occupational exposure: 1½ inch valves/4 connections; 2-3/35; 4/3; 6-8/11; 10-12/4; 14/1. Estimates of man hours to perform bolt replacements in the various connections are shown in Table 6.17. The low estimate is based on the assumption that some connections contain bolts, some connections contain studs, and that the task was performed smoothly. The best estimate was based upon the assumption that all connections contained studs, not bolts, a few minor problems were encountered in stud removal and replacement and some connections are changed without other work. The high estimate was based upon the assumption that all connections contained only studs, some problems were encountered in performing tasks (e.g., snapping a stud off or not being able to remove it and having to cut and drill out the stud and retap the female threads) and little other maintenance work is coordinated with the task. These estimates include data from EPRI NP-2174 (Looram et al. 1981), discussions with maintenance personnel from the Trojan Nuclear Plant, and engineering judgement.

The total occupational dose is calculated by multiplying the number of pertinent component connections by its associated dose rate and manhour estimate for bolt replacement. These data are added for each valve group component, then multiplied by the number of PWRs (47). Results are shown in Table 6.18.

TABLE 6.17. Manhour Estimates for Bolt Replacement per Connection

Components	Manhours - per - Connection		
	Low	Best	High
Pressurizer Manway Cover	1½	8	32
Steam Generator Manway Cover	2	10	40
Reactor Coolant Pump			
Case Flange	4	14	56
Seal Flange	1	5	20
Valve Body to Bonnet Connection (Valve Size-inches)			
1½	¼	2/3	3
2-3	¼	2/3	3
4	1/3	1½	6
6-8	1/3	1½	6
10-12	½	3	12
14	½	3	12

TABLE 6.18. Occupational Dose Results for PWRs

<u>Bolt Replacement Activity</u>	<u>Additional Occupational Dose (man-rem)</u>
Best Estimate	7.29
High Estimate	30.67
Low Estimate	1.99

6.3.2 BWR Occupational Exposure - Operational

For BWR occupational exposure, much of the same discussion and assumptions apply as for the PWR plants. The differences are in the number of components involved and the dose rates associated with replacing bolts or studs. The number of backfit BWRs affected is 24.

It is assumed that the recirculation pump internals will be serviced during the ten year inspection interval, and that bolt replacement can be accomplished during this maintenance work. Thus, no additional time for bolt replacement due to the proposed action is acquired. Also, bolt replacement on about two thirds of the valve body to bonnet connections in each valve group

of Table 6.20 are assumed to contribute to occupational exposure. The other bolt replacements are assumed to be performed when doing other maintenance or inservice inspection work.

Thus, for the nine valve component groups of Table 6.20 (i.e. 1½ inch thru 26 inches) the following number of connections are used for calculating occupational exposure: 1½ inch valves/3 connections; 2-3/5; 4/3; 6/9; 10-12/10; 14/6; 20/2; 24/9; 26/1.

The number of RCPB components with potential bolted connections was estimated from P&IDs and FSAR data of the WNP-2 nuclear power plant (WPPSS 1978). Components of concern included the recirculation pump case and seal flanges and valves greater than 1 in. diameter in the RCPB. Safety and relief valves for piping systems connected to the RCPB were not included. A description of RCPB valves used in this analysis is shown in Table 6.19. Bolted connection data used in this analysis are shown in Table 6.20. Dose rates and man-hour estimates for bolt removal and replacement are shown in Table 6.21. The BWR occupational dose increase due to this proposed action is shown in Table 6.22. The total additional occupational dose for all BWRs and PWRs is shown in Table 6.23.

TABLE 6.19. BWR RCPB Valve Description (Amount and Type) Nominal Valve Sizes (Inches)

<u>Valve Type</u>	<u>1½</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>6</u>	<u>10</u>	<u>12</u>	<u>14</u>	<u>20</u>	<u>24</u>	<u>26</u>
Check	2				3		6	3		4	
Gate	2		3	4	4	2	5	6	3	8	
Globe		4			2		2			2 ^(b)	2
Safety	—	—	—	—	5 ^(a)	—	—	—	—	—	—
TOTAL	4	4	3	4	14	2	13	9	3	14	2

(a) These safety valves have a 6 in. diameter inlet and a 10 in. diameter outlet.

(b) These were actually termed "flow control" valves.

TABLE 6.20. Bolted Connection Data for a BWR

<u>Components</u>	<u>Number of Connections</u>	<u>Number of Bolts or Studs per Connection</u>	<u>Size of Bolt or Stud (inches) (Diameter x Length)</u>
Recirculation Pump ^(a)			
Case Flange	2	20	3½ x 5
Seal Flange	2	20	1 x 5
Valve Body to Bonnet Connections (Valve Size-Inches)			
1½	4	4	½ x 2½
2-3	7	4	5/8 x 3
4	4	8	5/8 x 3½
6	14	8	3/4 x 4
10-12	15	12	7/8 x 4½
14	9	12	1 x 5
20	3	20	1-1/8 x 6
24	14	20	1-1/4 x 6-3/4
26	2	20	1½ x 7½

(a) These data were assumed to be the same as for the PWR reactor coolant pumps.

TABLE 6.21. BWR Dose Rates and Man-hour Estimates
for Bolt Replacement per Connection

Component	Dose Rate (mrem/hr)	Man-hour Estimates		
		Low	Best	High
Recirculation Pump				
Case Flange	200	4	14	56
Seal Flange	200	1	5	20
Valve Body to Bonnet Connections (Valve Size-Inches)				
1½	100	¼	2/3	3
2-3	100	¼	2/3	3
4	100	1/3	1½	6
6	100	1/3	1½	6
10-12	100	½	3	12
14	100	½	3	12
20	100	5/6	5	20
24	100	5/6	5	20
26	75	1	5½	22

TABLE 6.22. Occupational Dose Results for BWRs

<u>Bolt Replacement Activity</u>	<u>Additional Occupational Dose (man-rem)</u>
Best Estimate	313.10
High Estimate	1258.80
Low Estimate	57.40

TABLE 6.23. Total Increase in Occupational Dose for all Backfit Plants

<u>Bolt Replacement Activity</u>	<u>Additional Occupational Dose (man-rem)</u>
Best Estimate	320
High Estimate	1290
Low Estimate	60

6.4 PUBLIC PROPERTY

The effect of the proposed action on reducing the risk to public (offsite) property is calculated by multiplying the change in accident probability by a generic offsite property damage estimate. This estimate was derived from the mean value of results of CRAC 2 calculations, assuming an SST1 release (major accident) for 154 reactors (Strip 1982).

The following discounting formula is employed:

$$D = V \left[\frac{e^{-rt_i} - e^{-rt_f}}{r} \right]$$

where D = discounted value

V = damage estimate

t_i = years before reactor begins operation; 0 for operating reactor

t_f = years remaining until end of life

r = real discount rate

The damage estimate was converted to a present value by discounting it at a 10% rate. A 5% discount rate was also considered as a sensitivity case. For this proposed action, only backfit reactors are affected, and the average remaining lifetime is 26.9 years. Therefore, the 10% discount factor $D/V = 9.3$. The 5% discount factor equals 14.8. These values must be multiplied by the number of affected facilities (71) and the core-melt accident frequency to yield the total effect of the action. High and lower bounds are values for Indian Point 2 and Palo Verde 3 calculated from Strip (1982). The results are shown in Table 6.24.

6.5 ONSITE PROPERTY

The effect of the proposed action on reducing the risk to onsite property is estimated by multiplying the change in accident probability by a discounted onsite property cost. This discounted property cost was developed from the generic onsite property cost taken from Andrews et al. (1983). It includes an estimate for replacement power.

TABLE 6.24. Public Property Damage

	Offsite Property Damage (\$/event)	Discounted Offsite Property Damage [lifetime Risk] (\$/event)		Discounted Value of Additional Offsite Property Damage (\$)	
		5%	10%	5%	10%
Best Estimate	1.7E+09	2.5E+10	1.6E+10	3.5E+06	2.2E+06
High Estimate	9.2E+09	1.4E+11	8.6E+10	5.0E+07	3.1E+07
Low Estimate	8.3E+08	1.2E+10	7.7E+09	0.0	0.0

Property damage costs were discounted using the following formula.

$$D = \frac{V}{m} \left[\frac{e^{-rt_i}}{r^2} \right] [(1 - e^{-rm}) (1 - e^{-r(t_f - t_i)})]$$

where: D = discounted value

V = damage estimate

m = years over which cleanup is spread = 10 years for core-melt

t_i = years before reactor begins operation = 0 for operating plants

t_f = years remaining until end of life = 26.9 years

r = discount rate = 5% or 10%

For this proposed action, the 10% discount factor (D/V) equals 5.9, and the 5% discount factor equals 11.6. To obtain the total effect of the action, the per-reactor results are multiplied by the number of affected facilities (71) and the reduction in accident frequencies. The uncertainty bounds given in the table reflect a 50% spread which was estimated to be indicative of the uncertainty level. The results are summarized in Table 6.25.

6.6 INDUSTRY IMPLEMENTATION COSTS

An important aspect of replacing RCPB bolting with materials better resistant to corrosion is to assure that the material does indeed meet requested and required specifications. The benefits sought from replacing a high strength low alloy steel bolt with a specific type of stainless steel bolt would be degraded if the stainless steel bolt received from the manufacturer did not meet specified standards. There are currently no definitive rules for the receiving inspection of bolts at nuclear sites (Brookhaven National Laboratory 1984). The only requirements are general in nature. e.g., 10 CFR 50 App B -

TABLE 6.25. Onsite Property Damage

<u>Accident Type</u>	<u>Onsite Property Damage Estimate (\$/event)</u>	<u>Discounted Value of Avoided Onsite Property Damage (\$)</u>	
		<u>5%</u>	<u>10%</u>
Core Melt			
Best Estimate	1.6E+09	2.6E+06	1.3E+06
High Estimate	2.5E+09	1.1E+07	5.4E+06
Low Estimate	8.2E+08	0.0	0.0

Section VII, Control of Purchased Material, Equipment, and Services. It is basically up to the individual utility to develop a receiving inspection program for its purchased material. A SRP for RCPB bolting currently in revision offers guidelines on acceptance criteria for bolting material specifications and fabrications and processing of bolting materials.

Regarding studs being out of specification for hardness measurement, NUREG-0943 cited instances where either an improper heat treatment, fabrication quench cracks, or use of wrong material were contributing factors to bolt or stud premature failure. Bush and Simonen (1982) also cited incidents of failed bolting being out of specification at the Midland and Arizona Nuclear Power Project plants. If more extensive receiving inspections had been in effect, these problems could possibly have been avoided. The extremely small sample sizes used for qualifying huge lots of material is currently considered unacceptable for any bolts which would have a safety related function.

This analysis assumes an upgraded receiving inspection for bolting materials will be implemented at backfit plants, as part of the proposed action.

A possible intensified receiving inspection plan for bolting/fastener material could consist of:

1. A visual inspection of all incoming bolting material.
2. Hardness testing of 5 to 10% (at random) of the "as received" bolts/studs. If any fall outside the specified hardness range of the material, all bolts should either be returned to vendor or 100% hardness tested to eliminate non-conforming items.
3. A surface nondestructive examination (dye penetrant or magnetic particle) of a percentage (1 to 5%) of the "as received" bolts in order to preclude the possibility of pre-existing flaws in the bolts. This action is to supplement items 1 and 2.
4. Purchase of an hardness tester.

In all cases, the higher percentage of inspection would be used in the more critical bolting structures/installations.

A quality assurance or quality control organization could handle such testing. This analysis assumes the following implementation costs.

TABLE 6.26. Estimated Total Industry Implementation Costs

<u>Resources</u>	<u>Cost (\$)</u>		
	<u>Best Estimate</u>	<u>High Estimate</u>	<u>Low Estimate</u>
Equipment ^(a)	4.7E+04	1.4E+05	1.2E+04
Labor ^(b)	2.1E+05	8.5E+05	7.1E+04
Administration ^(c)	<u>2.1E+05</u>	<u>5.0E+05</u>	<u>9.4E+04</u>
TOTAL	4.7E+05	1.5E+6	1.8E+05

(a) Based on data from McMaster-Carr Supply Company catalog, 1985.

(b) Includes labor for performing bolt receiving, hardness testing, and stocking control.

(c) Includes resources for bolt purchasing and quality control and/or quality assurance.

6.7 INDUSTRY OPERATION COSTS

This analysis assumes that industry operation costs due to this proposed action will consist of the cost for stainless steel bolts or studs and the labor involved with bolt or stud replacement. It is assumed that no extension of outage time will occur and, thus, no replacement power costs will result. The amount and size of bolts or studs that would be purchased and the labor for replacement have been estimated in the section on operational occupational exposure. Labor cost is based upon rate of \$56.75 per man-hour as developed by Andrews et al. (1983). Cost estimates for 3½, 2, and 1½ in. diameter bolts or studs and nuts obtained from the ESCO Company (Spokane, Washington), were based on 304 stainless steel material. The prices obtained were estimated for hex head bolts and hex nuts. It is assumed that these same prices apply to studs. Cost for smaller diameter bolts or studs were based on information out of the 1985 McMaster-Carr Supply Company catalog. Bolt sizes and estimated costs are shown in Table 6.27. It was assumed that one nut per bolt or stud would be included in the cost for bolt replacement.

The total industry cost is a combination of the cost for bolt materials and the labor to replace bolts/studs. The material cost was found by multiplying the number of bolts/studs per connection times the number of connections for each component, times the cost for the corresponding size bolt (with nut). This is for all of the connections shown in Tables 6.15 and 6.20. These products were then added for each component to obtain the total cost of bolts or studs to be purchased per plant. These products were then multiplied by

TABLE 6.27. Costs for Stainless Steel Bolt/Studs

<u>Bolt Size Diameter x Length in inches</u>	<u>Bolt/Stud Cost \$ - Each</u>	<u>Nut Cost \$ - Each</u>	<u>Total Cost \$ - Each</u>
3½ x 5	200.00	122.00	322.00
2 x 5	34.00	33.00	67.00
1½ x 7½	20.00	7.00	27.00
1½ x 5	18.00	7.00	25.00
1½ x 6-3/4	14.00	5.00	19.00
1-1/8 x 6	11.00	4.00	15.00
1 x 5 ^(a)	8.30	2.31	10.61
7/8 x 4½ ^(a)	5.50	1.62	7.12
3/4 x 4 ^(a)	3.45	0.97	4.42
5/8 x 3½ ^(a)	2.25	0.56	2.81
5/8 x 3 ^(a)	2.00	0.56	2.56
½ x 2½ ^(a)	1.25	0.25	1.75

(a) Bolt/Stud costs were based on the cost of a broken package of 100 of type 303 stainless steel milled studs divided by 100. Nut costs were based on costs for 18-8 stainless steel.

the number of affected plants for the PWR and BWR case, then added together. The operational cost for labor was found by multiplying the total number of connections that had bolts replaced due to this proposed action (i.e. about two-thirds of valve body to bonnet connections), times the manhour estimate for bolt replacement for each component connection. In addition, labor times for bringing new SS bolts to the work area and discarding old bolts were estimated. It was assumed that for each connection, a low estimate of one-half manhour, a best estimate of one manhour, and affected a high estimate of two manhours would be added to the time for actual bolt replacement. These data were added up for each component and multiplied by the respective number of affected plants. These results were then added to obtain the total man-hours for operation of this action. This was then multiplied by the \$56.75/man-hour rate to obtain the total cost for labor. This labor cost (for each estimate case) was then added with the material cost to acquire the total industry cost for operation.

This analysis assumes that the proposed action will occur during the time for one ten year inservice inspection interval. The total costs shown in Table 6.28 are discounted using the following formula:

$$PV = \frac{A (1 + r)^n - 1}{r(1 + r)^n}$$

where PV = present value discounted

A = annual cost over affect period of operation (10 years)

r = interest rate (5% or 10%)

n = remaining period (10 years).

The results are shown in Table 6.28.

TABLE 6.28. Estimated Total 10 Year Industry Operation Costs in 1984 Dollars

	<u>Best Estimate</u>		<u>High Estimate</u>		<u>Low Estimate</u>	
PWR						
Materials	1.74E+06		1.74E+06		1.74E+06	
Labor	3.20E+05		1.01E+06		1.22E+05	
BWR						
Materials	5.70E+05		5.70E+05		5.70E+05	
Labor	2.45E+05		8.53E+05		6.56E+04	
TOTAL	2.88E+06		4.17E+06		2.50E+06	
	<u>5%</u>	<u>10%</u>	<u>5%</u>	<u>10%</u>	<u>5%</u>	<u>10%</u>
TOTAL (Discounted)	2.22E+06	1.77E+06	3.22E+06	2.56E+06	1.93E+06	1.54E+06

6.8 NRC DEVELOPMENT COSTS

A lot of attention and study on bolting problems in nuclear power plants has arisen over recent years. Revisions being incorporated into a new SRP for RCPB bolting reflect some of the results of these recent studies. An example is in the change in philosophy regarding bolt strength. Recent data has shown that applying a high accurate preload to a bolt of "lower strength" can provide greater resistance to SCC than use of a higher strength bolt. It has been found that "the stronger the bolt the better off you'll be" philosophy opens itself up to corrosion problems. Upper limits on bolting yield strengths are being suggested.

To establish this analysis' proposed action, it is assumed that NRC costs for developing criteria for a receiving/inspection program will occur. In addition, it is further assumed that the NRC will "pull together" results of available studies and develop preload and material specifications for stain-

less steel bolt replacement in RCPB applications. It is assumed that this development will entail a best estimate of one-half man year of effort with an assumption of \$100,000 per man-year. This results in an estimate of \$5.0E+04 for NRC development for this proposed action. A high estimate based on 3 times the best estimate, and a low estimate 1/3 the best estimate value was assumed. Results are shown below in Table 6.29.

TABLE 6.29. Estimated NRC Development Costs in 1984 Dollars

Best Estimate	5.0E+04
High Estimate	1.5E+05
Low Estimate	1.7E+04

6.9 NRC IMPLEMENTATION COSTS

This cost will result from the NRC putting the proposed action into effect. This may involve issuing a bulletin(s) to affected licensees concerning: criteria for an upgraded receiving/inspection program for bolting materials and stainless steel preload material specifications and their application to RCPB bolting applications. NRC implementation costs may also include resources for review and acceptance of a licensee's receiving/inspection program, or design change review and concurrence for bolt replacements in specific (critical) applications; or for concurrence on licensees' requests for not replacing bolts with corrosion resistant SS materials. Resources and cost estimates are shown below.

TABLE 6.30. NRC Implementation Costs

<u>Resources</u>	<u>Cost (\$)</u>		
	<u>Best Estimate</u>	<u>High Estimate</u>	<u>Low Estimate</u>
Document Preparation and Administration	9.0E+03	2.7E+04	3.0E+03
Review of Licensees' Implementation Efforts	<u>2.0E+04</u>	<u>6.0E+04</u>	<u>6.7E+03</u>
TOTAL	2.9E+04	8.7E+04	9.7E+03

7.0 CONCLUSIONS

Summary results for the value-impact assessment are shown in Table 7.1. The best estimate for cost and dose indicate that the proposed action has potential to reduce risk (9,819 man-rem) at a cost savings of about \$1.2E+06. The high estimate shows that risk can be reduced (25,036 man-rem) with a cost savings (\$3.2E+07). The low estimate indicates that public risk could increase at an increased cost to industry and the NRC. Based on this analysis, the proposed action can be recommended for implementation. Following is a summary of the value-impact assessment using a 10% discount rate.

TABLE 7.1. Summary of Value-Impact Assessment (10% Discount Rate)

Value (man-rem)			Impact (\$)		
Best Est.	High Est.	Low Est.	Best Est.	High Est.	Low Est.
9,819	25,036	-60	-1.2E+06	-3.2E+07	1.7E+06

Following is a summary of the value-impact assessment using a 5% discount rate.

TABLE 7.2. Summary of Value-Impact Assessment (5% Discount Rate)

Value (man-rem)			Impact (\$)		
Best Est.	High Est.	Low Est.	Best Est.	High Est.	Low Est.
9,819	25,036	-60	-3.3E+06	-5.6E+07	2.1E+06

As can be seen from Table 4.0 Value-Impact Assessment Summary, the averted onsite and public property damage cost savings control the total quantified impact for the best and high estimate cases. Exclusion of the averted onsite and offsite costs would increase the ratio of net implementation cost to man-rem averted. This is reflected below in Table 7.3 which excludes these two impacts.

TABLE 7.3. Summary of Value-Impact Assessment Excluding Averted Onsite and Offsite Property Costs (10% Discount Rate)

Value (man-rem)			Impact (\$)		
Best Est.	High Est.	Low Est.	Best Est.	High Est.	Low Est.
9,819	25,036	-60	2.3E+06	4.4E+06	1.7E+06

The proposed action of this value-impact assessment was to replace bolts in RCPB over a ten year interval. However, the public risk reduction developed in Section 6.1 and listed in Table 4.0 was not time averaged over this ten

year period. Time averaging this risk over a ten year interval would tend to lower the estimated risk reduction. Assuming that 10% of the needed bolt replacements were performed each year over the ten year period, starting at a remaining life of 26.9 years for all plants, a simple linear analysis shows that the estimated public risk reduction would be lowered by about 15%. Thus, a more accurate value for public health, to account for the ten year implementation schedule, would be approximately 8,346 man-rem for a best estimate and 21,281 man-rem for a high estimate.

This value-impact assessment was developed for operating reactors (ORs) (i.e. 47 PWRs and 24 BWRs). Note that the costs and risk would vary from ORs to forward-fit reactors with construction permits (CPs), operating licenses (OLs), and near-term operating licenses (NTOLs). For CPs and OLs, there would be no occupational exposure involved with implementing this proposed action. Also, depending upon the degree of plant completion, there may be little or no extra cost for buying and installing new bolts in RCPB connections. For NTOLs, both costs and occupational exposure would be reduced from that of ORs. Though a NTOL reactor may have loaded fuel, the connections for bolt replacement would have a reduced exposure rate than for an OR. Also, assembly/disassembly of bolted connections should be just as easy or easier than in ORs, due to less time for bolts to become gummed up or corroded. Since radiation exposure would be less, personnel could work longer times inside containment replacing bolts, which could reduce labor costs. For estimating occupational exposure and labor costs for a NTOL reactor due to this proposed action, one could use the low manhour estimates in Tables 6.17 and 6.21. These low estimates for an OR may be more typical for NTOL reactor bolt replacement.

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APPENDIX B

VALVE-IMPACT ANALYSIS
FOR BOLTS NOT LOCATED ON THE
REACTOR COOLANT PRESSURE BOUNDARY

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VALUE-IMPACT ANALYSIS
FOR BOLTS NOT LOCATED ON THE
REACTOR COOLANT PRESSURE BOUNDARY

1.0 INTRODUCTION

Generic Issue (GI) 29, "Bolting Degradation or Failure in Nuclear Power Plants", was issued as a result of an increase in the number of bolting-related incidents reported by the licensees of operating reactors and reactors under construction.¹ A large number of the reported bolting incidents have been related to primary pressure boundary applications and major component support structures. A Value-Impact Analysis (VIA) for replacing the reactor coolant pressure boundary (RCPB) bolts of BWRs and PWRs was prepared by PNL in 1985 under instructions limiting the VIA to the potential for failure of primary pressure boundary bolting². In addition, recent research has been sponsored by the AIF and EPRI for industry resolution of generic issue GI-29 and is reported in a two-volume draft report issued in June 1987³. The report presents results of the detailed investigations into the bolting issue and makes recommendations for resolution of the bolting problem.

This report presents the results of a VIA that INEL was requested to perform that includes non primary systems to be compatible with the resolution of the broader issue, as stated in a letter from Robert Baer, Chief of Engineering Issues Branch, to Joseph Murphy, Chief of Probabilistic Risk Analysis Branch⁴. By NRC-RES direction, the scope of the VIA is restricted to PWRs, internal bolts in primary system components are not within the scope of this VIA, and bolts covered by the previous VIA performed by PNL are not to be considered. Section 2 of this report presents the proposed action considered in this VIA; Sections 3, 4, and 5 provide listings of the affected decision factors, a value-impact assessment summary, and a statement of the unquantified assessment, respectively; quantification of the costs to implement the proposed action and the avoided public risk are developed in Section 6; and conclusions are presented in Section 7.

2.0 EVALUATION OF PROPOSED AND ALTERNATE ACTIONS

This section presents the proposed action and potential alternate actions that will minimize the potential for bolting problems in nuclear power plants. The proposed action was provided by NRC Staff personnel.

2.1 PROPOSED ACTION

Because of concern about increasing bolt degradation, the NRC is considering the need for providing specific guidance on bolting. The purpose of this is to prevent the two most common failure modes of bolts. These are stress corrosion cracking (SCC), which was identified in a 1983 NRC report⁵ as the most common mode of bolt failure in data previous to 1983, and borated water corrosion (BWC), which was the dominant failure mode of reactor coolant pressure boundary bolts from subsequent data and reported in a 1985 NRC report⁶.

The proposed regulatory approach for reducing bolting failures in non-RCPB bolting applications is to survey bolt documentation to determine if the installed bolts could be susceptible to either SCC or BWC, test bolts that may be susceptible to failure, and then replace those that do not meet the testing requirements. This approach is similar to that proposed by industry in the EPRI/AIF report³. Figure 1 shows a schematic of the proposed action.

The survey will include a review of records to determine the bolting material, expected material properties, the expected environment, preload, and application for each bolted joint that is included in the VIA. The survey process will be similar to that outlined in the Draft Standard Review Plan, "Reactor Bolting", Third Draft, November 11, 1987⁷. Based on the survey, joints with a bolting material that has experienced failures, bolts that could be susceptible to stress corrosion cracking (SCC), and bolted joints for which adequate redundancy has not been demonstrated to exist, will be subjected to testing. The testing will include hardness tests to determine actual material properties and inspection of the bolts to determine the actual condition of the bolts. Visual inspections are the only tests that are presently endorsed by ASME Section XI⁸ for detecting SCC in most bolts sizes found in component support applications. Therefore, it is anticipated that the testing of bolts will require removal, visual inspection, and reinstallation. Disposition of bolts that fail the testing will require replacement of the problem bolts.

Figure 1 Schematic of Proposed Action

2.2 POTENTIAL ALTERNATES

Existing field ultrasonic examination (UT) technology does not reliably detect borated water corrosion (wastage) of bolts, except for large diameter bolts that have heater holes, or the initiation of SCC. Therefore the use of improved inspection techniques and more frequent inspections may be a viable approach for identifying degraded bolts. The EPRI research³ investigated some improved inspection techniques for performing field NDE inspections to detect wastage and SCC in pressure boundary and component support bolting. One improved technique, Cylindrically Guided Wave Technique (CGWT), was identified and the study concluded that the use of CGWT could identify degraded bolts, as shown in the following quote:

"The cylindrically guided wave technique provided an inspection method applicable to most studs or bolts over a range of 16 to 112 inches (406 to 2,844 mm) in length and 1 to 4.5 inches (25.4 to 114 mm) in diameter. The technique can be used to detect cracklike defects as small as 0.05 inch (1.27 mm) deep in the threaded region of the bolt. In addition, the CGWT can be used to detect corrosion wastage greater than 25% of the bolt diameter."

However, improved techniques such as CGWT have not been incorporated in the ASME Section XI code⁸ and are not likely to have general acceptance in the short term. Therefore, the alternate action of utilizing improved inspection techniques has not been considered further.

Another alternate is to take no action. This alternate would not result in a decreased potential for bolt failures to cause a problem and is not considered further in this VIA. However, this alternate could become attractive in the future if it is determined that the potential for bolt failures to cause a safety problem is extremely low.

3.0 SELECTION AND EVALUATION OF PARAMETERS

"Parameters" considered in this study are shown in Table 3.1

Table 3.1 Checklist for Identification of Parameters

<u>Parameters considered</u>	<u>Causes a Quantified Change</u>	<u>Causes an Unquantified Change</u>	<u>No Change</u>
Public Health	X		
Occupational Exposure (Accidental)	X		
Occupational Exposure (Operational)	x		
Offsite Property	X		
Onsite Property	X		
Regulatory Efficiency		(a)	
Improvements in Knowledge		(a)	
Industry Implementation	X		
Industry Operation	X		
NRC Development			X
NRC Implementation	X		
NRC Operation			X

(a) The changes are not readily estimated in dollars and are not significant factors for evaluation of the this VIA.

4.0 VALUE-IMPACT ASSESSMENT SUMMARY

This section contains a summary of the estimates for the value-impact decision factors affected in this study. Industry costs, NRC costs, averted public risk, and the value-impact ratio are shown in Table 4.1. The supporting factors such as occupational exposures, averted public property costs, and averted onsite property costs are shown in Table 4.2.

Table 4.1 Value-Impact Assessment Summary - Total for all PWRs

<u>Decision Factors</u>	<u>Best Estimate</u>	<u>High Estimate</u>	<u>Low Estimate</u>
<u>Values (Averted man-rem)</u>			
Public Health	7.3E+03	1.4E+05	0
<u>Impacts (\$)</u>			
Industry Implementation Cost	6.0E+06	1.1E+07	3.0E+06
Industry Operating Cost	2.1E+07	2.6E+07	1.6E+07
NRC Costs	6.6E+04	1.2E+05	1.2E+04
<u>Value-Impact Ratio (\$/man-rem)</u>	3708	265	(a)

(a) Undefined because the ratio $1.9E+07/0 = \text{infinity}$.

Table 4.2 Supporting Factors - Total for all PWRs

<u>Decision Factors</u>	<u>Best Estimate</u>	<u>High Estimate</u>	<u>Low Estimate</u>
<u>Values (Averted man-rem)</u>			
Occupational Exposure (Accidental)	999.5	9058.3	0.0
Occupational Exposure (Operational)	10.4	43.7	2.8
<u>Impacts (\$)</u>			
Public Property	3.8E+07	1.8E+09	0.0
Onsite Property	2.1E+07	5.8E+08	0.0

5.0 UNQUANTIFIED PARAMETERS

There are no unquantified parameters that significantly affect the results of this study.

6.0 DETAILED EVALUATIONS

6.1 PUBLIC HEALTH

6.1.1 Selection of Systems

The initial list of systems and bolting applications that could be included in the VIA is quite large. Including them all in the VIA would likely result in analyzing some that have little, if any, effect on public risk. Therefore only those bolt failures that are significant contributors to public risk were considered. The following is a discussion of the factors that were considered in determining which systems and bolts are significant.

A review of bolt failure data reported in NUREG-0943⁵, and the research reported by EPRI³, show that the frequency of bolting failures is low. Therefore it is assumed that simultaneous failure of multiple systems is very unlikely and may be excluded from this analysis. Research sponsored by EPRI, "Degradation and Failure of Bolting in Nuclear Power Plants", Research Project 2520-7, June 1987, considered the frequency of degraded bolting in RCPB applications for each nuclear plant from criticality to September 30, 1984. There were 796 degraded bolts out of 31569 bolts at risk, in a total of 52 nuclear plants accumulating about 477.5 plant years of operation since initial criticality. This corresponds to a degradation rate of 1.67 per plant year (0.28% of the bolts at risk in a plant per year).

The study of bolt failure events, sponsored by NRC and reported in NUREG-0943, "Threaded-Fastener Experience in Nuclear Power Plants", showed that events usually involved more than one bolt failure, although the exact number was not always stated. This study provides a list of the reported bolt failure events from 1964 to March 1982. A total of 43 events are reported, 39 in PWRs and the remainder (4) in BWRs. Eleven PWR component support events were reported for 47 PWRs accumulating about 350 plant years of operation. This corresponds to 0.031 events per plant year. If each event involved 10 failed or degraded bolts the rate of bolt failures would be 0.31 failures per plant year. (Note that the low number of bolt failures reported in BWRs supports the NRC direction to limit this VIA to PWRs.)

The rate of bolt failures, or degradation, reported by EPRI for RCPB applications (1.67 rejects/py) appears to be conservative for all bolting applications. Because most bolted joints involve several bolts, this low rate of bolt failures supports the conclusion that simultaneous bolted joint failures in multiple systems may be excluded from this analysis.

Research reported by EPRI³ supports the conclusion that bolted pressure boundary joints will fail with a leak-before-break sequence permitting failures to be detected and fixed prior to a catastrophic failure that initiates a transient within the plant. Preliminary analyses of several bolted RCPB joints were performed to assess a leak-before-break strategy. The preliminary analyses considered the following closures:

- 16 stud manway cover
- 20 stud manway cover
- Reactor coolant pump flange (16 studs)
- 6-inch check valve (12 studs)
- 10-inch check valve (16 studs)

Results of the analyses show that at 1 GPM leakage a computed safety margin of 2.2 to 3.2 exists with 7.8% (pump) to 27.7% (6-inch valve) of the studs in one component failed. Since the number of failed studs is expected to be small, as previously discussed, and the leak rate is sufficient to be detected, it is reasonable to assume that degraded pressure boundary joints will be identified by a noticeable leak and repaired before a catastrophic failure initiates a reactor system transient or the affected system becomes unavailable to perform its function.

Because of the leak-before-break assumption, failures of pressure boundary bolts are not likely to initiate serious or significant accident sequences. Other initiating events such as water hammers and earthquakes may result in sufficient stresses to cause leakage at degraded pressure boundary bolt connections. But because the number of degraded bolts is expected to be small and a significant number of bolts must be degraded at a single bolted joint for the joint to fail, the leaking bolted pressure boundary joints will not cause systems to fail to function. It was concluded that pressure boundary bolts need not be considered further in this VIA.

Water hammers are not produced in service water, component cooling water, residual heat removal, low pressure injection, high pressure injection, and charging systems during normal and accident operations because they are always filled with single phase water. A study of water hammers, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants", March 1984⁹, was performed to resolve Generic Safety Issue A-1, "Water Hammer". The report identified 67 known or suspected water hammer events in PWRs from 1969 to 1981. Feedwater water hammer including those in the feedring of the steam generator accounted for 59.7% of the total while occurrences in the main steam system and reactor primary system accounted for 11.9% and 7.5% respectively. There were no reported water hammer events involving the diesel generators and so they are not considered in this study. Therefore, water hammers will be considered for only the main steam, feedwater, and auxiliary feedwater systems.

NUREG-0943 reported 6 events involving internal bolting of PWR components with all but 2 being in the reactor vessel. One of the 2 involved a main steam isolation valve stud and the other involved a service water pump impeller capscrew. Because the reported failure rate of internal bolts in components outside the primary system is very low and internal bolting of primary system component is not within the scope of this study, internal bolting will not be included in this VIA.

Degradation of bolts in component supports is not expected to initiate accident sequences but is important for accident sequences that are initiated by events that significantly stress the support systems, such as water hammers or seismic events. While Large LOCAs in the primary system also stress component supports, PWRs have multiple primary loops and a LOCA in one loop will not result in large stresses to component supports in the unbroken loops. Even though component supports in the broken loop are stressed and could fail, the loop is already broken and added failures will not increase the severity of the accident. Therefore, only water hammers and seismic events are considered to be significant sources of stress for component supports.

In summary, the failure of pressure boundary bolts is not expected to result in the initiation of accident sequences and is not expected to result in failure of system function during accident sequences that involve water hammer or earthquake initiators. Water hammers are not expected to be a factor in many systems but are considered for the main feedwater, auxiliary feedwater, and main steam systems. Degradation of component supports is not expected to initiate accident sequences but is important for seismic and water hammer events. Table 6.1 shows the systems and bolting applications that will be considered further in this study.

6.1.2 Development of Hybrid PRA

A hybrid probabilistic risk assessment (PRA) model was developed to evaluate the changes in core damage frequency caused by bolting failures in the systems of interest. The need for the hybrid PRA was determined after review of several PRA's which included seismic risk in the total contribution to plant damage. No one model contained accident sequences which included all of the systems of interest in this VIA. Rather than limit the number of systems included, a decision was made to develop a hybrid plant model which would contain representative seismic core damage sequences for at least a majority of the systems of interest.

This model was constructed after review of six PRA's which included seismic risk in the total core damage frequency. The plants reviewed were:

- Zion
- Indian Point 2
- Indian Point 3
- Seabrook
- Millstone 3
- Oconee

The total core melt frequency and the frequency associated with seismic events for each of these plants is presented in Table 6.2.

TABLE 6.1 SELECTION OF SYSTEMS AND BOLTS
FOR BOUNDING CALCULATIONS

System	Bolting Application		
	Pressure Boundary	Component Supports	Component Internals
Primary System	N (1)	SEISMIC	N(2)
Secondary to Isol.	N	WH, SEISMIC	N
Main Feedwater	N	WH, SEISMIC	N
Auxiliary FW	N	WH, SEISMIC	N
HPI	N	SEISMIC	N
LPI	N	SEISMIC	N
Accumulator Inject	N	SEISMIC	N
Charging System	N	SEISMIC	N
Service Water	N	SEISMIC	N
Component Cooling	N	SEISMIC	N
Diesel	N	SEISMIC	N

Notes: 1. Reactor coolant pressure boundary bolts studies by PNL
2. By NRC direction

WH - Water Hammer could be an initiator
Seismic - Seismic events could be an initiator
N - Not considered further in the VIA

Table 6.2 Core Melt Frequency and Seismic Contribution^a

<u>Plant</u>	<u>Total Core Melt</u>	<u>Seismic Contribution</u>
Indian Point 2	1.4E-04	7.7E-06
Indian Point 3	1.4E-04	3.6E-06
Zion	5.4E-05	5.6E-06
Seabrook	2.3E-04	2.0E-05
Millstone3	5.0E-05	9.1E-06
Oconee	7.5E-05	2.1E-05

- (a) The seismic hazard curves from the Seabrook Station Probabilistic Safety Assessment were used to determine the frequency of exceedance values for the seismic initiating events. The results would be similar for a PWR located in the Midwest or east coast.

Plants on the west coast can experience earthquakes of larger magnitude and therefore are designed to a higher earthquake acceleration. Because the components are designed for higher accelerations, the results of this VIA would be similar if a west coast site were considered.

The procedure followed in developing this hybrid model is as follows:

- 1) Identify the dominant accident scenarios from each PRA.
- 2) Identify and study the sequences initiated by seismic events, focusing on those seismic failures caused by component bolting, either support or pressure boundary.
- 3) Assemble a specialized plant model which contains representative dominant sequences from the plant reviewed.
- 4) Assemble sequences from the six PRA's that cover the systems of interest to the resolution of this VIA.
- 5) Quantify this list of dominant sequences using typical (generic) values for component failures, initiating event frequencies, and seismic fragilities.
- 6) Establish a "representative" core damage frequency from the hybrid model, e.g. baseline the model.

The sequences developed for the hybrid model are identified in Table 6.3. The baseline core damage frequency from this model is 3.1E-04 per reactor year. The seismic sequences included in the model contribute approximately 10% (3.0E-05 per reactor year) to the total core damage frequency. These core damage frequencies are higher than those listed for the six PRA's, Table 6.2, because the hybrid PRA is a compilation of all the systems that are in the individual PRA's. None of the individual PRA's include all the systems that are included in the hybrid PRA. However the seismic contribution of the hybrid PRA (10%) is nearly identical to the average of the seismic contributions of the six PRA's (12%).

Table 6.3 RESULTS OF QUANTIFICATION – BOUNDING CASE

	Base	AFW	HP1	LPI	CCW	EPS/SWS	STEAM	WATER
Core Damage Total1	3.08E-04	3.23E-04	3.21E-04	3.20E-04	3.19E-04	6.02E-04	3.13E-04	3.08E-04
LOSP * GA * GB * ER	3.97E-05	----	----	----	----	----	----	----
RTRIP * PA * PB	2.11E-05	----	----	----	----	----	----	----
PLMF * PA * PB	1.71E-05	----	----	----	----	----	----	----
SLOCA * L1 * L2	1.69E-05	----	----	----	----	----	----	----
TTRIP * PA * PB	1.32E-05	----	----	----	----	----	----	----
RTRIP * SA * SB * OR	1.17E-05	----	----	----	----	----	----	----
LIDC * EF * FR * FR	1.14E-05	----	----	----	----	----	----	----
SLBI * ON	9.39E-06	----	----	----	----	1.40E-05	----	----
EXFW * PA * PB	9.34E-06	----	----	----	----	----	----	----
FCRCC	9.00E-06	----	----	----	----	----	----	----
TTRIP * SA * SB * PR	7.29E-06	----	----	----	----	----	----	----
LOMF * SA * SB	5.69E-06	----	----	----	----	----	----	6.01E-06
E0.7T * OG * GA * GB	5.65E-06	----	----	----	1.86E-05	----	----	----
SLOCA * EA * L2	5.55E-06	----	----	----	----	----	----	----
SLOCA * EB * L1	5.55E-06	----	----	----	----	----	----	----
LOSP * PA * PB	4.96E-06	----	----	----	----	----	----	----
FPCC	4.20E-06	----	----	----	----	----	----	----
LOPF * PA * PB	3.79E-06	----	----	----	----	----	----	----
FSRCC	3.60E-06	----	----	----	----	----	----	----
SLOCA * WA * L2	3.26E-06	----	----	----	----	----	----	----
SLOCA * WB * L1	3.26E-06	----	----	----	----	----	----	----
RTRIP * WA * WB * SR	3.14E-06	----	----	----	----	----	----	----
LOSP * GA * GB * EF * ER * FR	3.01E-06	----	----	----	----	----	----	----
LOSP * WA * WB * ER	2.57E-06	----	----	----	----	----	----	----
PLMF * WA * WB * SR	2.54E-06	----	----	----	----	----	----	2.54E-06
LOSW	2.52E-06	----	----	----	----	----	----	----
SLOCA * WB * EA	2.41E-06	----	----	----	----	----	----	----
SLOCA * WA * EB	2.41E-06	----	----	----	----	----	----	----
FTBLP * GA * GB * ER	2.33E-06	----	----	----	----	----	----	----
LOSP * GA * GB * EF * ER	2.33E-06	----	----	----	----	----	----	----
E0.5T * OG * GA * GB	2.32E-06	----	----	----	1.84E-05	----	----	----
LOSP * GA * WB * ER	2.12E-06	----	----	----	----	----	----	----
LOSP * GB * WA * ER	2.12E-06	----	----	----	----	----	----	----
RTRIP * ON	2.11E-06	----	----	----	----	----	----	----
E0.4T * OG * GA * GB	2.08E-06	----	----	----	3.60E-05	----	----	----
TTRIP * WA * WB * SR	1.95E-06	----	----	----	----	----	----	----

Table 6.3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
V SEQUENCE	1.84E-06	----	----	----	----	----	----	----
PLMFW * ON	1.71E-06	----	----	----	----	----	----	----
FCRSW	1.65E-06	----	----	----	----	----	----	----
E0.7T * OG * SA * SB * PA * PC	1.63E-06	----	----	----	5.85E-06	0.00E+00	----	----
FCRAC	1.62E-06	----	----	----	----	----	----	----
FLSW	1.60E-06	----	----	----	----	----	----	----
E0.7T * OG * GA * GB * EK	1.59E-06	2.85E-06	----	----	----	5.26E-06	----	----
E1.0T * OG * GA * GB	1.25E-06	----	----	----	----	2.16E-06	----	----
LLOCA * LA * LB	1.14E-06	----	----	----	----	----	----	----
E0.2T * OG * GA * GB	1.09E-06	----	----	----	----	1.27E-04	----	----
EXFW * ON	9.34E-07	----	----	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----	----	----
RTRIP * EF * O3	8.32E-07	----	----	----	----	----	----	----
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	----	2.34E-06	0.00E+00	0.00E+00	----	----
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	2.50E-06	0.00E+00	----	0.00E+00	----	----
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	----	----	----	5.43E-06	----	6.97E-07
LOMF * RT * OH	6.59E-07	----	----	----	----	----	----	----
E1.0L * OG * GA * GB	6.52E-07	----	----	----	----	1.12E-06	----	----
LOSP * WA * WB * EF * ER * FR	6.34E-07	----	----	----	----	----	----	----
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	----	4.26E-06	----	0.00E+00	----	----
E0.7L * OG * SA * SC	5.57E-07	----	----	----	----	0.00E+00	----	----
E0.7L * OG * GA * GB	5.38E-07	----	----	----	----	1.77E-06	----	----
TTRIP * EF * L1 * L2	5.20E-07	----	----	----	----	----	----	----
TTRIP * EF * O3	5.18E-07	----	----	----	----	----	----	----
E0.5T * OG * EJ * H2	5.13E-07	1.51E-06	3.67E-06	0.00E+00	----	0.00E+00	----	----
E0.4T * OG * GA * GB * EI	5.06E-07	1.83E-06	----	----	----	8.73E-06	----	----
FSRAC	4.90E-07	----	----	----	----	----	----	----
E0.4T * OG * EI * H2	4.53E-07	1.96E-06	6.47E-06	0.00E+00	----	0.00E+00	----	----
TTRIP * RT * OH	4.49E-07	----	----	----	----	----	----	----
E0.7T * OG * SA * SB * EK * H2	4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00	----	----
E0.5T * OG * SA * SB * PA * PB	4.24E-07	----	----	----	3.53E-06	0.00E+00	----	----
E0.4T * OG * EI * RW	3.83E-07	1.66E-06	----	6.38E-06	----	0.00E+00	----	----
LOPF * ON	3.79E-07	----	----	----	----	----	----	----
LOSP * DA * EF * ER * FR	3.75E-07	----	----	----	----	----	----	----
LOSP * DB * EF * ER * FR	3.75E-07	----	----	----	----	----	----	----

Table 6.3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
EXFW * EF * L1 * L2	3.68E-07	----	----	----	----	----	----	----
EXFW * EF * O3	3.66E-07	----	----	----	----	----	----	----
E1.0L * OG * SA * SC	3.31E-07	----	----	----	----	0.00E+00	----	----
LOSP * WA * WB * ON * ER	3.28E-07	----	----	----	----	----	----	----
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	----	4.75E-07	----	0.00E+00	----	----
LCV * ON	2.83E-07	----	----	----	----	----	----	----
SLBI * L1 * L2	2.73E-07	----	----	----	----	4.09E-07	----	----
SLBI * O3	2.72E-07	----	----	----	----	4.08E-07	----	----
ELOCA	2.66E-07	----	----	----	----	----	----	----
SGTR * EF * O3	2.62E-07	----	----	----	----	----	----	----
MLOCA * L1 * L2	2.54E-07	----	----	----	----	----	----	----
RTRIP * OG * GA * GB * ER	2.44E-07	----	----	----	----	----	----	----
CMSIV* ON	2.39E-07	----	----	----	----	----	----	----
LOSP * GA * WB * EF * ER * FR	2.31E-07	----	----	----	----	----	----	----
LOSP * GB * WA * EF * ER * FR	2.31E-07	----	----	----	----	----	----	----
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	----	----	----	3.88E-07	----	3.36E-07
TLMFW * ON	2.24E-07	----	----	1.17E-06	0.00E+00	0.00E+00	----	----
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	----	----	----	----	----	----
LCV * EF * O3	1.52E-07	----	----	----	----	----	----	----
TTRIP * OG * GA * GB * ER	1.52E-07	----	----	----	----	----	----	----
LOSP * WA * WB * EF * ER	1.51E-07	----	----	----	----	----	----	----
LOSP * DA * GB * ER	1.48E-07	----	----	----	----	----	----	----
LOSP * DB * GA * ER	1.48E-07	----	----	----	----	----	----	----
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00	----	----
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	----	0.00E+00	1.15E-06	0.00E+00	----	----
LOSP * GB * WA * EF * ER	1.25E-07	----	----	----	----	----	----	----
LOSP * GA * WB * EF * ER	1.25E-07	----	----	----	----	----	----	----
LOSP * GA * WB * ON * ER	1.19E-07	----	----	----	----	----	----	----
LOSP * GB * WA * ON * ER	1.19E-07	----	----	----	----	----	----	----
EXFW * OG * GA * GB * ER	1.07E-07	----	----	----	----	----	----	----
E0.2T * OG * GA * GB * EG	9.56E-08	1.09E-06	----	----	----	1.10E-05	----	----
MLOCA * EB * L2	9.22E-08	----	----	----	----	----	----	----
MLOCA * EA * L1	9.22E-08	----	----	----	----	----	----	----
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	----	0.00E+00	----	3.02E-07	----	----
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	----	0.00E+00	----	1.40E-07	----	----
LOSP * WB * EF * OR * ER * FR	6.94E-08	----	----	----	----	----	----	----
LOSP * WA * EF * OR * ER * FR	6.94E-08	----	----	----	----	----	----	----

Table 6.3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
SLOCA * WA * WB * SR	6.82E-08	---	---	---	---	---	---	---
E0.2T * OG * GB * WA	6.26E-08	---	---	---	---	0.00E+00	---	---
E0.2T * OG * GA * WB	6.26E-08	---	---	---	---	0.00E+00	---	---
LLOCA * LC * LD	6.12E-08	---	---	---	---	---	---	---
E0.2T * OG * WA * WB	5.47E-08	---	---	---	---	0.00E+00	---	---
LOSP * DA * GB * EF * FR * ER	5.07E-08	---	---	---	---	---	---	---
LOSP * DB * GA * EF * FR * ER	5.07E-08	---	---	---	---	---	---	---
LLOCA * EA * LB	3.60E-08	---	---	---	---	---	---	---
LLOCA * EB * LA	3.60E-08	---	---	---	---	---	---	---
SGTR * WA * WB * SR	3.26E-08	---	---	---	---	---	---	---
LOSP * WA * PB * ER	2.87E-08	---	---	---	---	---	---	---
LOSP * WB * PA * ER	2.87E-08	---	---	---	---	---	---	---
LOSP * DA * WB * ER	2.15E-08	---	---	---	---	---	---	---
LOSP * DB * WA * ER	2.15E-08	---	---	---	---	---	---	---
E0.3T * SA * SB * EH * C2	2.03E-08	1.25E-07	---	---	---	---	---	---
MLOCA * WB * L1	1.96E-08	---	---	---	---	---	---	---
MLOCA * WA * L2	1.96E-08	---	---	---	---	---	---	---
E0.4T * SA * SB * E1 * C2	1.90E-08	6.90E-08	---	---	---	---	---	---
L1DC * PA * EF * FR * FR	7.35E-09	---	---	---	---	---	---	---
L1DC * PB * EF * FR * FR	7.35E-09	---	---	---	---	---	---	---
SLOCA * L1 * L2 * XA	5.94E-09	---	---	---	---	---	---	---
E0.4T * SA * SB * RW * C2	3.93E-09	---	---	6.56E-08	0.00E+00	---	---	---
E0.5T * SA * SB * RW * C2	3.36E-09	---	---	2.40E-08	0.00E+00	---	---	---
E0.3T * SA * SB * H3 * C2	2.64E-09	---	1.25E-07	---	---	---	---	---
E0.3T * SA * SB * RW * C2	2.47E-09	---	---	1.23E-07	0.00E+00	---	---	---
Other Sequences Not Included in Model	8.00E-06	---	---	---	---	---	---	---
Core Damage Due to Seismic Events								
E0.7T * OG * GA * GB	2.92E-05	4.39E-05	4.23E-05	4.18E-05	3.99E-05	3.23E-04	---	---
E0.5T * OG * GA * GB	5.65E-06	---	---	---	---	1.86E-05	---	---
E0.4T * OG * GA * GB	2.32E-06	---	---	---	---	1.84E-05	---	---
E0.7T * OG * SA * SB * PA * PC	2.08E-06	---	---	---	---	3.60E-05	---	---
E0.7T * OG * SA * SB * GB * EK	1.63E-06	---	---	---	5.85E-06	0.00E+00	---	---
E0.3T * SA * SB	1.59E-06	2.85E-06	---	---	---	5.26E-06	---	---
E0.3T * OG * GA * GB	1.54E-06	---	---	---	---	---	---	---
E0.7T * OG * EK * RW	1.34E-06	---	---	---	---	7.34E-05	---	---
E1.0T * OG * GA * GB	1.28E-06	2.43E-06	---	3.79E-06	---	0.00E+00	---	---
	1.25E-06	---	---	---	---	2.16E-06	---	---

Table 6.3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HP1	LPI	CCW	EPS/SWS
E0.2T * OG * GA * GB	1.09E-06	-----	----	----	----	1.27E-04
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	-----	2.34E-06	0.00E+00	0.00E+00
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	2.50E-06	0.00E+00	-----	0.00E+00
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	-----	-----	-----	5.43E-06
E1.0L * OG * GA * GB	6.52E-07	-----	-----	-----	6.52E-07	1.12E-06
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	-----	4.26E-06	-----	0.00E+00
E0.7L * OG * SA * SC	5.57E-07	-----	-----	-----	-----	0.00E+00
E0.7L * OG * GA * GB	5.38E-07	-----	-----	-----	-----	1.77E-06
E0.5T * OG * EJ * H2	5.13E-07	1.51E-06	3.67E-06	0.00E+00	-----	0.00E+00
E0.4T * OG * GA * GB * EI	5.06E-07	1.83E-06	-----	-----	-----	8.73E-06
E0.4T * OG * EI * H2	4.53E-07	1.96E-06	6.47E-06	0.00E+00	-----	0.00E+00
E0.7T * OG * SA * SB * EK * H2	4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB	4.24E-07	-----	-----	-----	3.53E-06	0.00E+00
E0.4T * OG * EI * RW	3.83E-07	1.66E-06	-----	6.38E-06	-----	0.00E+00
E1.0L * OG * SA * SC	3.31E-07	-----	-----	-----	-----	0.00E+00
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	-----	4.75E-07	-----	0.00E+00
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	-----	-----	-----	3.88E-07
E0.3T * OG * GA * GB * EH	2.09E-07	1.29E-06	-----	-----	-----	1.14E-05
E0.4T * OG * SA * SB * PA * PB	2.03E-07	-----	-----	-----	4.07E-06	0.00E+00
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	-----	1.17E-06	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	-----	0.00E+00	1.15E-06	0.00E+00
E0.2T * OG * GA * GB * EG	9.56E-08	1.09E-06	-----	-----	-----	1.10E-05
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	-----	0.00E+00	-----	3.02E-07
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	-----	0.00E+00	-----	1.40E-07
E0.2T * OG * GB * WA	6.26E-08	-----	-----	-----	-----	0.00E+00
E0.2T * OG * GA * WB	6.26E-08	-----	-----	-----	-----	0.00E+00
E0.2T * OG * WA * WB	5.47E-08	-----	-----	-----	-----	0.00E+00
E0.3T * SA * SB * EH * C2	2.03E-08	1.25E-07	-----	-----	-----	-----
E0.4T * SA * SB * EI * C2	1.90E-08	6.90E-08	-----	-----	-----	-----
E0.4T * SA * SB * RW * C2	3.93E-09	-----	-----	6.56E-08	0.00E+00	-----
E0.5T * SA * SB * RW * C2	3.36E-09	-----	-----	2.40E-08	0.00E+00	-----
E0.3T * SA * SB * H3 * C2	2.64E-09	-----	1.25E-07	-----	-----	-----
E0.3T * SA * SB * RL * C2	2.47E-09	-----	-----	1.23E-07	0.00E+00	-----

6.1.3 Bounding studies

Bounding calculations have been performed using the hybrid model for sequences that include seismic and water hammer initiating events and for the systems that were listed for consideration. For the bounding calculations, the affected systems were assumed to fail with a probability of one, given a seismic initiating event having a magnitude equal to the design basis earthquake. As can be seen from Table 6.4, failures of most systems changed total core melt by less than 5%. Failure of the diesel generators or the service water system resulted in 50% change in total core damage. This change in core melt frequency is essentially the same as the frequency of the seismic event. These results indicate that the electric power system and its support systems play a significant role in the prevention of core damage after a seismic initiating event. This conclusion is not surprising when one realizes that components in the offsite power distribution system typically are more fragile than the components at a nuclear power plant. Failure in the Reactor Coolant System (LOCA) resulting from seismic events has the same effect on core damage as failure of the electric power system. This is due primarily to the assumption that more than one loop will be affected during the seismic event and the assumption that any seismic event fails the system. Core damage is guaranteed if more than one loop fails.

The systems which were identified as being susceptible to water hammer failure were quantified under bounding conditions similar to the systems affected by the seismic events. The initiating frequency for total loss of main feedwater was increased by 50%, from 0.33 events per reactor year to 0.50 events per reactor year, and the initiating event frequency for steam line breaks was increased 50%, from 6.0E-03 to 9.0E-3 events per reactor year to model an increase in bolt failures given a water hammer event. The results shown below indicate that even with this increase, large changes in core damage frequency are not seen and bolting failures as a result of water hammer are not a problem.

WATER HAMMER EVENTS

<u>System</u>	<u>Core Damage</u>	
	<u>Total</u>	<u>Change</u>
FEEDWATER	3.1E-04	3.2E-06
STEAM	3.1E-04	5.0E-06

6.1.4 Analysis of Risk

The bounding calculations described previously indicate that this analysis should focus on the likelihood of system bolt failure (component supports, pressure boundary, etc.) during a seismic event. The assumption of system failure due to bolting failure for any earthquake at or above the design

Table 6.4 RESULTS OF BOUNDING CALCULATIONS INCLUDING
BOLT FAILURES DURING SEISMIC EVENTS

<u>System</u>	<u>Core Damage Total</u>	<u>Frequency Change</u>
Base Case	3.1E-04	---
Failure of:		
- AFW	3.2E-04	1.5E-05
- HPI	3.2E-04	1.3E-05
- LPI	3.2E-04	1.3E-05
- CCW	3.2E-04	1.1E-05
- EPS / SWS	6.0E-04	2.9E-04
- LOCA	6.0E-04	2.9E-04
- ACCUMULATOR	No Change	(1)
- CHARGING	No Change	(2)
- STEAM	No Change	(3)
- FEEDWATER	No Change	(4)

- (1) Failure of the accumulators during a seismic event is assumed not to cause an initiating event directly. Because the Large LOCA initiated by seismic failure is assumed to cause failure of multiple RCS loops, success or failure of the accumulators does not affect core damage likelihood.
- (2) The charging system is included with failure of the HPI system for the purposes of this analysis. Failure of the normal charging function during seismic events does not significantly affect the likelihood of core damage.
- (3) An increase in the likelihood of failure of the steam lines downstream of the MSIV's during a seismic event has no effect on the sequences modeled. The frequency of core damage is dominated by the failures in the other plant systems.
- (4) An increase in the likelihood of failure of the feedwater system during a seismic event has no effect on the sequences modeled. The frequency of core damage is dominated by the failures in the other plant systems.

basis earthquake acceleration is felt to be extremely conservative and so the bounding calculations are used as the high estimate. For the purpose of evaluating this VIA, a more realistic quantification is necessary. Since there are no data to support a change in component failure frequency due to bolt degradation, engineering judgement is used to determine a best estimate likelihood of the effects of degraded bolting on core damage frequency. For the purpose of calculating the best estimate, it is assumed that, if degraded component support or pressure boundary bolting exists at twice the design basis earthquake acceleration, component failure would be guaranteed. The plant model was requantified using this assumption. This assumption is similar to that used in NUREG-0577, Rev. 1¹⁰ which investigated degraded pump and steam generator support failures during seismic events. The results of the quantification using this assumption are shown below.

RESULTS OF BEST ESTIMATE CALCULATIONS
SEISMIC FAILURE

System	Core Damage Total	Core Damage Change
Base Case	3.08E-04	-----
AFW	3.17E-04	8.9E-06
HPI	3.15E-04	7.0E-06
LPI	3.15E-04	6.9E-06
CCW	3.19E-04	1.1E-05
EPS / SWS	3.40E-04	3.2E-05
LOCA	3.40E-04	3.2E-05

Under the assumption described above, the maximum change in core damage frequency due to failure of degraded bolts at twice design basis earthquake accelerations is seen in the electric power or reactor coolant systems and the change is approximately 10% of the total core damage frequency. Because of the plant damage sequences that are included in the hybrid model, the maximum change in core damage frequency is approximately equal to the frequency of exceeding an earthquake acceleration of twice design basis.

Table 6.5 lists the changes in sequence frequency under the best case assumptions. Table 6.3 lists the changes in sequence frequency under high estimate assumptions. Table 6.6 describes the top event coding used in Tables 6.3 and 6.5. The use of Table 6.6 is described in the following example.

The second sequence in Table 6.3 and Table 6.5 is: LOSP * GA * GB * ER. The first entry is the initiating event, in this case a loss of offsite power. The remaining entries describe subsequent system/top event failures in the hybrid model. Referring to Table 6.6,

GA	Diesel Generator Train A fails
GB	Diesel Generator Train B fails
ER	Electric Power System Recovery Models

Table 6.5 Change in Core Melt Frequency - Best Case

	Base	AFW	HPI	LPI	CCW	EPS/SWS
Core Damage Total	3.08E-04	3.17E-04	3.15E-04	3.15E-04	3.19E-04	3.40E-04
LOSP * GA * GB * ER	3.97E-05	----	----	----	----	----
RTRIP * PA * PB	2.11E-05	----	----	----	----	----
PLMFW * PA * PB	1.71E-05	----	----	----	----	----
SLOCA * L1 * L2	1.69E-05	----	----	----	----	----
TTRIP * PA * PB	1.32E-05	----	----	----	----	----
RTRIP * SA * SB * OR	1.17E-05	----	----	----	----	----
LIDC * EF * FR * FR	1.14E-05	----	----	----	----	----
SLBI * ON	9.39E-06	----	----	----	----	----
EXFW * PA * PB	9.34E-06	----	----	----	----	----
FCRCC	9.00E-06	----	----	----	----	----
TTRIP * SA * SB * PR	7.29E-06	----	----	----	----	----
LOMF * SA * SB	5.69E-06	----	----	----	----	----
E0.7T * OG * GA * GB	5.65E-06	----	----	----	----	1.86E-05
SLOCA * EA * L2	5.55E-06	----	----	----	----	----
SLOCA * EB * L1	5.55E-06	----	----	----	----	----
LOSP * PA * PB	4.96E-06	----	----	----	----	----
FPCC	4.20E-06	----	----	----	----	----
LOPF * PA * PB	3.79E-06	----	----	----	----	----
FSRCC	3.60E-06	----	----	----	----	----
SLOCA * WA * L2	3.26E-06	----	----	----	----	----
SLOCA * WB * L1	3.26E-06	----	----	----	----	----
RTRIP * WA * WB * SR	3.14E-06	----	----	----	----	----
LOSP * GA * GB * EF * ER * FR	3.01E-06	----	----	----	----	----
LOSP * WA * WB * ER	2.57E-06	----	----	----	----	----
PLMFW * WA * WB * SR	2.54E-06	----	----	----	----	----
LOSW	2.52E-06	----	----	----	----	----
SLOCA * WB * EA	2.41E-06	----	----	----	----	----
SLOCA * WA * EB	2.41E-06	----	----	----	----	----
FTBLP * GA * GB * ER	2.33E-06	----	----	----	----	----
LOSP * GA * GB * EF * ER	2.33E-06	----	----	----	----	1.84E-05
E0.5T * OG * GA * GB	2.32E-06	----	----	----	----	----
LOSP * GA * WB * ER	2.12E-06	----	----	----	----	----
LOSP * GB * WA * ER	2.12E-06	----	----	----	----	----
RTRIP * ON	2.11E-06	----	----	----	----	----
E0.4T * OG * GA * GB	2.08E-06	----	----	----	----	----

Table 6.5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
TTRIP * WA * WB * SR	1.95E-06	----	----	----	----	----
V SEQUENCE	1.84E-06	----	----	----	----	----
PLMFW * ON	1.71E-06	----	----	----	----	----
FCRSW	1.65E-06	----	----	----	5.85E-06	0.00E+00
E0.7T * OG * SA * SB * PA * PC	1.63E-06	----	----	----	----	----
FCRAC	1.62E-06	----	----	----	----	----
FLSW	1.60E-06	2.85E-06	----	----	----	5.26E-06
E0.7T * OG * GA * GB * EK	1.59E-06	----	----	----	----	----
LOSP * GA * GB * ON * ER	1.56E-06	----	----	----	----	----
E0.3T * SA * SB	1.54E-06	----	----	----	----	----
FLLP * GA * GB * ER	1.43E-06	----	----	----	----	----
EXFW * WA * WB * SR	1.38E-06	----	----	----	----	----
LPCC	1.38E-06	----	----	----	----	----
E0.3T * CG * GA * GB	1.34E-06	----	----	----	----	----
TTRIP * ON	1.32E-06	2.43E-06	----	3.79E-06	----	0.00E+00
E0.7T * OG * EK * RW	1.28E-06	----	----	----	----	2.16E-06
TCTL * GA * GB * ER	1.28E-06	----	----	----	----	----
E1.0T * OG * GA * GB	1.25E-06	----	----	----	----	----
LLOCA * LA * LB	1.14E-06	----	----	----	----	----
E0.2T * OG * GA * GB	1.09E-06	----	----	----	----	----
EXFW * ON	9.34E-07	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----
RTRIP * EF * 03	8.32E-07	----	----	----	----	----
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	2.50E-06	2.34E-06	0.00E+00	0.00E+00
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	----	0.00E+00	----	0.00E+00
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	----	----	----	5.43E-06
LOMF * RT * OH	6.59E-07	----	----	----	----	----
E1.0L * OG * GA * GB	6.52E-07	----	----	----	----	1.12E-06
LOSP * WA * WB * EF * ER * FR	6.34E-07	----	----	4.26E-06	----	0.00E+00
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	----	----	----	0.00E+00
E0.7L * OG * SA * SC	5.57E-07	----	----	----	----	1.77E-06
E0.7L * OG * GA * GB	5.38E-07	----	----	----	----	----
TTRIP * EF * L1 * L2	5.20E-07	----	----	----	----	----
TTRIP * EF * 03	5.18E-07	----	----	----	----	----

Table 6.5 Change in Core Melt Frequency - Best Case (Cont'd)

		Base	AFW	HPI	LPI	CCW	EPS/SWS
E0.5T * OG * EJ * H2		5.13E-07	1.51E-06	3.67E-06	0.00E+00	---	0.00E+00
E0.4T * OG * GA * GB * EI		5.06E-07	6.96E-07	---	---	---	---
FSRAC		4.90E-07	---	---	---	---	---
E0.4T * OG * EI * H2		4.53E-07	6.69E-07	---	---	---	---
TTRIP * RT * OH		4.49E-07	---	---	---	---	---
E0.7T * OG * SA * SB * EK * H2		4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB		4.24E-07	---	---	3.53E-06	---	---
E0.4T * OG * EI * RW		3.83E-07	5.66E-07	---	---	---	0.00E+00
LOPF * ON		3.79E-07	---	---	---	---	---
LOSP * DA * EF * ER * FR		3.75E-07	---	---	---	---	---
LOSP * DB * EF * ER * FR		3.75E-07	---	---	---	---	---
EXFW * EF * L1 * L2		3.68E-07	---	---	---	---	---
EXFW * EF * O3		3.66E-07	---	---	---	---	---
E1.0L * OG * SA * SC		3.31E-07	---	---	---	---	0.00E+00
LOSP * WA * WB * ON * ER		3.28E-07	---	---	---	---	---
E1.0T * OG * SA * SB * EL * RW		2.95E-07	3.95E-07	---	4.75E-07	---	0.00E+00
LCV * ON		2.83E-07	---	---	---	---	---
SLBI * L1 * L2		2.73E-07	---	---	---	---	---
SLBI * O3		2.72E-07	---	---	---	---	---
ELOCA		2.66E-07	---	---	---	---	---
SGTR * EF * OD		2.62E-07	---	---	---	---	---
MLOCA * L1 * L2		2.54E-07	---	---	---	---	---
RTRIP * OG * GA * GB * ER		2.44E-07	---	---	---	---	---
CIMSIV * ON		2.39E-07	---	---	---	---	---
LOSP * GA * WB * EF * ER * FR		2.31E-07	---	---	---	---	---
LOSP * GB * WA * EF * ER * FR		2.31E-07	---	---	---	---	---
E1.0T * OG * GA * GB * RT * EL		2.24E-07	3.01E-07	---	---	---	3.88E-07
TLMFW * ON		2.24E-07	---	---	---	---	---
LOSP * GA * EF * OR * ER * FR		2.10E-07	---	---	---	---	---
LOSP * GB * EF * OR * ER * FR		2.10E-07	---	---	---	---	---
RTRIP * EF * OR * FR * FR		2.09E-07	---	---	---	---	---
E0.3T * OG * GA * GB * EH		2.09E-07	---	---	---	---	---
E0.4T * OG * SA * SB * PA * PB		2.03E-07	---	---	---	4.07E-06	---
LOSP * EF * OR * ER * FR		2.02E-07	---	---	---	---	---
LOSP * GA * PB * ER		1.98E-07	---	---	---	---	---
LOSP * GB * PA * ER		1.98E-07	---	---	---	---	---

Table 6.5 Change in Core Melt Frequency - Best Case (Cont'd)

		Base	AFW	HPI	LPI	CCW	EPS/SWS
PLMFW	* OG * GA * GB * ER	1.97E-07	---	---	---	---	---
E0.5T	* OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	---	1.17E-06	0.00E+00	0.00E+00
LCV	* EF * O3	1.52E-07	---	---	---	---	---
TTRIP	* OG * GA * GB * ER	1.52E-07	---	---	---	---	---
LOSP	* WA * WB * EF * ER	1.51E-07	---	---	---	---	---
LOSP	* DA * GB * ER	1.48E-07	---	---	---	---	---
LOSP	* DB * GA * ER	1.48E-07	---	---	---	---	---
E0.5T	* OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T	* OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	---	0.00E+00	1.15E-06	0.00E+00
LOSP	* GB * WA * EF * ER	1.25E-07	---	---	---	---	---
LOSP	* GA * WB * EF * ER	1.25E-07	---	---	---	---	---
LOSP	* GA * WB * ON * ER	1.19E-07	---	---	---	---	---
LOSP	* GB * WA * ON * ER	1.19E-07	---	---	---	---	---
EXFW	* OG * GA * GB * ER	1.07E-07	---	---	---	---	---
E0.2T	* OG * GA * GB * EG	9.56E-08	---	---	---	---	---
MLOCA	* EB * L2	9.22E-08	---	---	---	---	---
MLOCA	* EA * L1	9.22E-08	---	---	---	---	---
E0.7T	* OG * GA * GB * RT * EK	9.15E-08	1.64E-07	---	0.00E+00	---	3.02E-07
E1.0T	* OG * GA * GB * EL	8.11E-08	1.08E-07	---	0.00E+00	---	1.40E-07
LOSP	* WB * EF * OR * ER * FR	6.94E-08	---	---	---	---	---
LOSP	* WA * EF * OR * ER * FR	6.94E-08	---	---	---	---	---
SLOCA	* WA * WB * SR	6.82E-08	---	---	---	---	---
E0.2T	* OG * GB * WA	6.26E-08	---	---	---	---	---
E0.2T	* OG * GA * WB	6.26E-08	---	---	---	---	---
LLOCA	* LC * LD	6.12E-08	---	---	---	---	---
E0.2T	* OG * WA * WB	5.47E-08	---	---	---	---	---
LOSP	* DA * GB * EF * FR * ER	5.07E-08	---	---	---	---	---
LOSP	* DB * GA * EF * FR * ER	5.07E-08	---	---	---	---	---
LLOCA	* EA * LB	3.60E-08	---	---	---	---	---
LLOCA	* EB * LA	3.60E-08	---	---	---	---	---
SGTR	* WA * WB * SR	3.26E-08	---	---	---	---	---
LOSP	* WA * PB * ER	2.87E-08	---	---	---	---	---
LOSP	* WB * PA * ER	2.87E-08	---	---	---	---	---
LOSP	* DA * WB * ER	2.15E-08	---	---	---	---	---
LOSP	* DB * WA * ER	2.15E-08	---	---	---	---	---
E0.3T	* SA * SB * EH * C2	2.03E-08	---	---	---	---	---

Table 6.5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
MLOCA * WB * L1	1.96E-08	----	----	----	----	----
MLOCA * WA * L2	1.96E-08	----	----	----	----	----
E0.4T * SA * SB * EI * C2	1.90E-08	2.61E-08	----	----	----	----
L1DC * PA * EF * FR * FR	7.35E-09	----	----	----	----	----
L1DC * PB * EF * FR * FR	7.35E-09	----	----	----	----	----
SLOCA * L1 * L2 * XA	5.94E-09	----	----	----	----	----
E0.4T * SA * SB * RW * C2	3.93E-09	----	----	2.40E-08	0.00E+00	----
E0.5T * SA * SB * RW * C2	3.36E-09	----	----	----	0.00E+00	----
E0.3T * SA * SB * H3 * C2	2.64E-09	----	----	----	----	----
E0.3T * SA * SB * RW * C2	2.47E-09	----	----	----	0.00E+00	----
Other Sequences Not Included in Model	8.00E-06	----	----	----	----	----
Core Damage Due to Seismic Events						
E0.7T * OG * GA * GB	2.92E-05	3.81E-05	3.62E-05	3.61E-05	3.99E-05	6.14E-05
E0.5T * OG * GA * GB	5.65E-06	----	----	----	----	1.86E-05
E0.4T * OG * GA * GB	2.32E-06	----	----	----	----	1.84E-05
E0.7T * OG * SA * SB * PA * PC	2.08E-06	----	----	----	----	----
E0.7T * OG * GA * GB * EK	1.63E-06	----	----	----	5.85E-06	0.00E+00
E0.3T * SA * SB	1.59E-06	2.85E-06	----	----	----	5.26E-06
E0.3T * OG * GA * GB	1.54E-06	----	----	----	----	----
E0.7T * OG * EK * RW	1.34E-06	----	----	----	----	----
E1.0T * OG * GA * GB	1.28E-06	2.43E-06	----	3.79E-06	----	0.00E+00
E0.2T * OG * GA * GB	1.25E-06	----	----	----	----	2.16E-06
E0.7T * OG * SA * SB * EK * RW	1.09E-06	----	----	----	----	----
E0.7T * OG * EK * H2	7.98E-07	1.43E-06	----	2.34E-06	0.00E+00	0.00E+00
E0.5T * OG * GA * GB * EJ	7.00E-07	1.32E-06	2.50E-06	0.00E+00	----	0.00E+00
E1.0L * OG * GA * GB	6.86E-07	1.81E-06	----	----	----	5.43E-06
E0.5T * OG * EJ * RW	6.52E-07	----	----	----	----	1.12E-06
E0.7L * OG * SA * SC	5.97E-07	1.75E-06	----	4.26E-06	----	0.00E+00
E0.7L * OG * GA * GB	5.57E-07	----	----	----	----	0.00E+00
E0.5T * OG * EJ * H2	5.38E-07	----	----	----	----	1.77E-06
E0.4T * OG * GA * GB * EI	5.13E-07	1.51E-06	3.67E-06	0.00E+00	----	0.00E+00
E0.4T * OG * EI * H2	5.06E-07	6.96E-07	----	----	----	----
E0.7T * OG * SA * SB * EK * H2	4.53E-07	6.69E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB	4.34E-07	7.78E-07	----	----	3.53E-06	0.00E+00
	4.24E-07	----	----	----	----	----

Table 6.5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
E0.4T * OG * EI * RW	3.83E-07	5.66E-07	---	---	---	0.00E+00
E1.0L * OG * SA * SC	3.31E-07	---	---	---	---	0.00E+00
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	---	---	---	0.00E+00
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	---	4.75E-07	---	0.00E+00
E0.3T * OG * GA * GB * EH	2.09E-07	---	---	---	---	3.88E-07
E0.4T * OG * SA * SB * PA * PB	2.03E-07	---	---	---	4.07E-06	---
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	---	1.17E-06	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	---	0.00E+00	1.15E-06	0.00E+00
E0.2T * OG * GA * GB * EG	9.56E-08	---	---	---	---	---
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	---	0.00E+00	---	3.02E-07
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	---	0.00E+00	---	1.40E-07
E0.2T * OG * GB * WA	6.26E-08	---	---	---	---	---
E0.2T * OG * GA * WB	6.26E-08	---	---	---	---	---
E0.2T * OG * WA * WB	5.47E-08	---	---	---	---	---
E0.3T * SA * SB * EH * C2	2.03E-08	---	---	---	---	---
E0.4T * SA * SB * EI * C2	1.90E-08	2.61E-08	---	---	---	---
E0.4T * SA * SB * RW * C2	3.93E-09	---	---	---	0.00E+00	---
E0.5T * SA * SB * RW * C2	3.36E-09	---	---	2.40E-08	0.00E+00	---
E0.3T * SA * SB * H3 * C2	2.64E-09	---	---	---	---	---
E0.3T * SA * SB * RW * C2	2.47E-09	---	---	---	0.00E+00	---

Table 6.6 Top Event Coding

<u>INITIATING EVENTS</u>		
C1MSIV	Closure of One MSIV	
E0.2T	Seismic Initiating Event - 0.2g	(3.6E-04)
E0.3T	Seismic Initiating Event - 0.3g	(1.1E-04)
E0.4T	Seismic Initiating Event - 0.4g	(4.3E-05)
E0.5T	Seismic Initiating Event - 0.5g	(2.0E-05)
E0.7L	Seismic Initiating Event - LLOCA, 0.7g	(1.8E-06)
E0.7T	Seismic Initiating Event - General Transient, 0.7g	(1.9E-05)
E1.0L	Seismic Initiating Event - LLOCA, 1.0g	(1.1E-06)
E1.0T	Seismic Initiating Event - General Transient, 1.0g	(2.2E-06)
ELOCA	Excessive LOCA	
EXFW	Excessive Feedwater Flow	
FCRAC	Fire, Control Room - Loss of Power	
FCRCC	Fire, Control Room - Loss of PCC	
FCRSW	Fire, Control Room - Loss of Service Water	
FLLP	Flood - LOSP, Turbine Building	
FLSW	External Flood - Service Water	
FPCC	Fire, PCC Area	
FSRAC	Fire, Spreading Room - Loss of AC Power	
FSRCC	Fire, Cable Spreading Room - Loss of PCC	
FTBLP	Fire, Turbine Building	
L1DC	Loss of One DC Bus	
LCV	Loss of Condenser Vacuum	
LLOCA	Large LOCA	
LOMF	Loss of Main Feedwater (Total and Partial)	
LOPF	Loss of Primary Flow	
LOSP	Loss of Offsite Power	
LOSW	Loss of Service Water	
LPCC	Loss of Primary Component Cooling	
MLOCA	Medium LOCA	
PLMFW	Partial Loss of Main Feedwater	
RTRIP	Reactor Trip	
SGTR	Steam Generator Tube Rupture	
SLBI	Steam Line Break Inside Containment	
SLOCA1	Small LOCA (nonisolable)	
SLOCA2	Small LOCA (isolable)	
TCTL	Truck Crash into Transmission Line	
TLMFW	Total Loss of Main Feedwater	
TTRIP	Turbine Trip	
V	Interfacing Systems LOCA	

Table 6.6 (cont'd)

SYSTEM EVENT TREE TOP EVENTS

C2	Containment Purge Isolation
DA	DC Power Train A
DB	DC Power Train B
EA	Engineered Safety Features Actuation System (ESFAS) Train A
EB	Engineered Safety Features Actuation System (ESFAS) Train B
EF	Auxiliary Feedwater System
EG	Auxiliary Feedwater System - Earthquake 0.2g
EH	Auxiliary Feedwater System - Earthquake 0.3g
EI	Auxiliary Feedwater System - Earthquake 0.4g
EJ	Auxiliary Feedwater System - Earthquake 0.5g
EK	Auxiliary Feedwater System - Earthquake 0.7g
EL	Auxiliary Feedwater System - Earthquake 1.0g
ER	Electric Power System Recovery Models
FR	Auxiliary Feedwater Recovery
GA	Diesel Generator Train A
GB	Diesel Generator Train B
H2	High Pressure Injection (HPI) for SLOCA, etc.
H3	High Pressure Injection (HPI) for ATWS Events
L1	Low Pressure Injection (LPI) Train A - MLOCA Miniflow
L2	Low Pressure Injection (LPI) Train B - MLOCA Miniflow
LA	Low Pressure Injection (LPI) Train A - LLOCA
LB	Low Pressure Injection (LPI) Train B - LLOCA
LC	Low Pressure Recirculation (LPR) Train A
LD	Low Pressure Recirculation (LPR) Train B
O3	Operator Action - LPR or HPR
OD	Operator Action to Depressurize Steam Generators
OG	Electric Power Systems - Offsite Grid
OH	Operator Action - Manual Reactor Shutdown, ATWS
ON	Operator Action - Plant Stabilization
OR	Operator Action - Feed and Bleed, SGTR Break Flow
PA	Primary Component Cooling Train A
PB	Primary Component Cooling Train B
PR	PORV's in Feed and Bleed
RT	Reactor Protection System
RW	Refueling Water Storage Tank (RWST)
SA	Solid State Protection System (SSPS) - Train A
SB	Solid State Protection System (SSPS) - Train B
SC	Main Steam Functions - Steam Generator Relief
SR	Service Water System Recovery
WA	Service Water System - Train A
WB	Service Water System - Train B
XA	Containment Spray Recirculation Train A

The sequence listed is a loss of offsite power followed by failure of diesel generator A and diesel generator B with failure to recover offsite or onsite power before core damage occurs. The remainder of the sequences in Tables 6.3 and 6.5 are read in a similar manner.

For the remainder of the analyses performed to support this VIA, the change in core damage frequency associated with the LOCA initiating event will be used as a measure of the risk associated with degraded bolting. This decision is made for the following reason:

The LOCA case was analyzed under the assumption that all RCS loops would be affected by degraded bolting (e.g. perfect coupling). Under this assumption, failure of the RCS is guaranteed to result in core damage because of the inability to inject water into the vessel. Thus the change in core damage frequency can be directly related to initiating event frequency, thereby simplifying the calculations.

There are now three separate calculations of core damage frequency for this VIA. The benchmark calculation is taken to be the base case and assigned to the 'LOW' category for the purposes of evaluating this bolting issue. The first bounding calculation for the LOCA initiating event is assumed to be the high estimate case. The best estimate case is the best estimate described above for the LOCA initiating event. The following table summarizes the results for the low, best, and high estimates of core damage frequency for the bolting VIA.

	<u>Core Damage Frequency</u>	<u>Change From Base Case</u>
Low	3.08E-04	0.00
Best Estimate	3.40E-04	3.2E-05
High Estimate	6.03E-04	2.9E-04

The change in man-rem dose due to the calculated changes in core damage frequency must be calculated in order to quantify this VIA. This section describes the procedure used to calculate the change in man-rem associated with the change in core damage frequency for the best estimate and high estimate assumptions.

One plant, Seabrook, was selected as representative of the response of the containment and site to selected initiating events and plant damage states. The Seabrook Station Probabilistic Safety Assessment (SSPSA), PLG-0300,¹¹ describes in several tables the mapping of core damage by initiating event and/or plant damage state through containment response to the release categories defined for Seabrook. For the purposes of this analysis, it is assumed that the containment response given a particular initiating event and plant damage state is not affected by changes in plant damage state frequency. The site response given a particular release category from the containment is also assumed to be unaffected by the calculated changes in core damage frequency from this analysis. Under these assumptions, given a change in plant damage state frequency, the change in risk to the public can be determined by calculating the change in plant damage state frequency and multiplying this change by the appropriate response 'vector'.

The following tables from the SSPSA were used to develop the reduced matrix that was used to evaluate the change in man-rem for this VIA.

Table 13.1-3 presents the definitions of the release categories used in the SSPSA risk models.

Table 12.5-85 presents the man-rem associated with each release category.

Table 13.2-5a presents a matrix of the frequency of occurrence of each release category as a result of each initiating event.

Table 13.2-5b presents the frequency of occurrence of accident sequences grouped by initiating event for each release category.

These tables are included as Appendix A to this report.

The reduced matrix contains only the elements associated with the seismic events. All other entries are assumed to remain the same (e.g. not affected by this VIA). The base case matrix is presented as Table 6.7 and presents the Base Case man-rem associated with seismic events for each release category and the total man-rem. Man-rem is calculated by multiplying the release category frequency by the conversion listed in Table 12.5-85, appendix A, and the summing across all release categories. The base case man-rem calculated by this method is 223 man-rem.

The high estimate analysis man-rem calculation is presented in Table 6.8. The change in release category frequency for each earthquake level is determined and the results summed to obtain the total change in man-rem. The change at each element is determined as follows.

The change in core damage frequency for each earthquake level is calculated by subtracting the base case core damage frequency for each earthquake level from the initiating event frequency for each earthquake level.

The ratio of the base case release fraction to the total release frequency for each release category is determined for each element.

The change in core damage frequency is multiplied by the release fraction ratio for each earthquake level.

The change in each element is summed for each release category and presented as a change in release category frequency.

The high estimate man-rem calculated by this method is 2364 per reactor. The averted man-rem is the difference between the high estimate and the base case, $2364 - 223 = 2141$ averted man-rem (high estimate). The total averted high estimate is $1.43\text{E}+05$ man-rem for 67 operating reactors.

The best estimate calculation is performed in a manner similar to the high estimate analysis. The best estimate calculation is presented in Table 6.9. The results for the best estimate calculation are 332 man-rem per reactor and the averted man-rem is $332 - 223 = 109$ man-rem. The total best estimate for 67 operating reactors is $7.30\text{E}+03$ man-rem.

Table 6.7 Calculation of Man-Rem - Base Case

	S1	S2	S3	S5	S6	S1B	S2B	S3B	S5H	S2BV	S3BV	S4BV	S6BV	CHANGE	TOTAL	FREQ.
E0.7L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.25E-07
E1.0L	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.22E-07
E0.2T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.76E-06
E0.3T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.45E-06
E04.7	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.08E-06
E0.5T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.41E-06
E0.7T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.27E-06
E1.0T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.45E-06
E0.2A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.00E-10
E0.3A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.69E-06
E04.A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.15E-07
E05.A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.54E-07
E0.7A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.38E-06
E1.0A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.68E-07
CHANGE	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
SUM	3.57E-11	8.58E-09	5.32E-07	5.96E-05	7.31E-11	2.78E-10	3.15E-07	5.84E-05	3.00E-11	1.79E-05	8.03E-05	1.00E-05	2.43E-06			
MAN-REM	0.000	0.046	0.393	0.128	0.000	0.007	1.689	43.089	0.000	96.302	40.908	7.398	33.335		223.295	

Table 6.8 Man-Rem Calculation - High Estimate

	S1	S2	S3	S5	S6	S10	S28	S38	S5H	S2BV	S3BV	S4BV	S6BV	CHANGE	TOTAL	FREQ.
E0.7L	3.30E-12	1.36E-11	1.06E-09	9.60E-08	8.34E-15	1.73E-11	2.49E-10	1.34E-07	4.85E-13	6.48E-07	1.48E-07	9.87E-08	5.14E-08	1.18E-06	1.80E-06	6.25E-07
E1.0L	2.34E-13	9.62E-13	7.49E-11	6.80E-09	6.10E-16	3.42E-12	3.40E-11	1.63E-08	3.43E-14	4.13E-07	3.13E-08	2.09E-08	1.68E-08	5.05E-07	1.13E-06	6.22E-07
E0.2T	1.99E-10	8.38E-09	1.70E-06	6.34E-05	4.23E-12	1.42E-11	2.51E-07	1.25E-05	3.19E-10	2.81E-04	1.92E-07	2.44E-08	9.02E-10	3.59E-04	3.61E-04	1.76E-06
E0.3T	7.51E-11	3.15E-09	4.14E-07	2.40E-05	1.71E-12	2.16E-11	8.29E-07	1.47E-05	1.21E-13	6.70E-05	2.42E-06	3.00E-07	4.86E-08	1.10E-04	1.12E-04	2.45E-06
E04.1	2.05E-11	8.61E-10	8.77E-08	6.56E-06	7.17E-13	1.11E-11	5.08E-07	6.80E-06	3.30E-11	2.33E-05	1.60E-06	1.98E-07	6.58E-08	3.91E-05	4.32E-05	4.08E-06
E0.5T	5.91E-12	2.48E-10	2.25E-08	1.89E-06	2.90E-13	5.04E-12	2.44E-07	2.56E-06	9.51E-12	9.54E-06	9.90E-07	1.23E-07	1.46E-07	1.55E-05	1.99E-05	4.41E-06
E0.7T	1.64E-12	6.88E-11	5.77E-09	5.25E-07	1.27E-13	2.64E-12	1.63E-07	9.17E-08	2.64E-12	7.05E-06	7.25E-07	8.96E-08	1.91E-07	9.66E-06	1.89E-05	9.27E-06
E1.0T	2.19E-14	9.20E-13	8.12E-12	7.02E-09	2.68E-15	9.71E-14	7.00E-09	1.85E-08	3.53E-14	6.35E-07	3.42E-08	4.23E-09	1.62E-08	7.22E-07	2.17E-06	1.45E-06
E0.2A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			5.00E-10
E0.3A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			1.69E-06
E04.A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			3.15E-07
E05.A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			3.54E-07
E0.7A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			1.38E-06
E1.0A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			5.68E-07
CHANGE	3.06E-10	1.27E-08	2.23E-06	9.65E-05	7.09E-12	7.52E-11	2.00E-06	3.69E-05	3.65E-10	3.89E-04	6.14E-06	8.58E-07	5.37E-07			
SUM	3.42E-10	2.13E-08	2.76E-06	1.56E-04	8.02E-11	3.53E-10	2.32E-06	9.52E-05	3.95E-10	4.07E-04	8.64E-05	1.09E-05	2.97E-06		2,364.063	
MAN-REM	0.003	0.114	2.037	0.334	0.000	0.009	12.437	70.305	0.000	2,186.061	44.037	8.031	40.693			

Table 6.9 Man-Rem Calculation - Best Estimate

	S1	S2	S3	S5	S6	S1B	S2B	S3B	SSH	S2BV	S3BV	S4BV	S6BV	CHANGE	TOTAL	FREQ.
E0.7L	3.30E-12	1.36E-11	1.06E-09	9.60E-08	8.34E-15	1.73E-11	2.49E-10	1.34E-07	4.85E-13	6.48E-07	1.48E-07	9.87E-08	5.14E-08	1.18E-06	1.80E-06	6.25E-07
E1.0L	2.34E-13	9.62E-13	7.49E-11	6.80E-09	6.10E-16	3.19E-12	3.00E-11	1.63E-08	3.43E-14	4.13E-07	3.13E-08	2.09E-08	1.64E-08	5.05E-07	1.13E-06	6.22E-07
E0.2T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			1.76E-06
E0.3T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			2.45E-06
E0.4.T	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			4.08E-06
E0.5T	5.91E-12	2.48E-10	2.25E-08	1.89E-06	2.90E-13	5.04E-12	2.44E-07	2.56E-06	9.51E-12	9.54E-06	9.90E-07	1.23E-07	1.46E-07	1.55E-05	1.99E-05	4.41E-06
E0.7T	1.64E-12	6.88E-11	5.77E-09	5.25E-07	1.27E-13	2.64E-12	1.63E-07	9.17E-08	2.64E-12	7.05E-06	7.25E-07	8.96E-07	1.91E-07	9.64E-06	1.89E-05	9.27E-06
E1.0T	2.19E-14	9.20E-13	8.12E-12	7.02E-09	2.68E-15	9.71E-14	7.00E-09	1.85E-08	3.53E-14	6.35E-07	3.42E-08	4.23E-09	1.62E-08	7.22E-07	2.17E-06	1.45E-06
E0.2A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			5.00E-10
E0.3A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			1.69E-06
E0.4.A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			3.15E-07
E0.5.A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			3.54E-07
E0.7A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			1.38E-06
E1.0A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00			5.68E-07
CHANGE	1.11E-11	3.32E-10	2.94E-08	2.53E-06	4.29E-13	2.83E-11	4.14E-07	2.82E-06	1.27E-11	1.83E-05	1.93E-06	3.36E-07	4.22E-07			
SUM	4.68E-11	8.91E-09	5.61E-07	6.22E-05	7.36E-11	3.06E-10	7.29E-07	6.12E-05	4.27E-11	3.62E-05	8.22E-05	1.04E-05	2.86E-06		332.781	
MAN-REM	0.000	0.048	0.414	0.133	0.000	0.008	3.912	45.172	0.000	194.443	41.891	7.645	39.114			

6.2 OCCUPATIONAL EXPOSURE - ACCIDENTAL

From NUREG/CR-3568, "A Handbook for Value-Impact Assessment",¹² and the VIA performed by PNL² occupational exposure from accidents can be estimated as the product of the change in total core-melt frequency and the occupational exposure likely to occur in the case of a major accident. The change in core melt frequency has been estimated as $2.9\text{E}-04$ and $3.2\text{E}-05$ for the high and best estimates respectively.

The occupational exposure in the event of a major accident has two components. The first is the "immediate" exposure to the personnel onsite during the span of the event and its short term control. The second part is the longer term exposure involved with the cleanup and recovery from the accident.

The total avoided occupational exposure is calculated as follows:

$$D_{TOA} = NT\Delta F(D_{IO} + D_{LTO})$$

where:

D_{TOA} = total avoided occupational dose

N = number of affect facilities (67 operating¹¹)

T = average remaining lifetime (22.2 yr.¹¹)

ΔF = change in core-melt probability

D_{IO} = "immediate" occupational dose

D_{LTO} = long-term occupational dose

Results of the calculations are shown in Table 6.10. Estimates for immediate and long-term occupational dose were obtained from the VIA performed by PNL².

Table 6.10

	Reduction in Core Melt Frequency (events/py)	Immediate Occupational Dose (man-rem/event)	Long-term Occupational Dose (man-rem)	Total Avoided Occupational Exposure (man-rem)
Best Estimate	$3.2\text{E}-05$	$1\text{E}+03$	$2\text{E}+04$	999.5
High Estimate	$2.9\text{E}-04$	$4\text{E}+03$	$3\text{E}+04$	9058.3
Low Estimate	0		$1\text{E}+04$	0

6.3 OCCUPATIONAL EXPOSURE - OPERATIONAL

Occupational exposure will result from performing testing and replacements of some bolted connections, particularly those that are located inside the containment and are in close proximity to reactor coolant. The additional occupational exposure estimated by PNL for replacing the RCPB bolts in 47 PWRs was estimated to be 30.67 man-rem for the high estimate, 7.29 man-rem for the best estimate, and 1.99 man-rem for the low estimate. These values are not directly applicable to the bolts of this VIA, but they should provide an approximate indication of the occupational exposure that would be expected. The bolts covered by this VIA that are located in containment are not as near to the reactor coolant piping as the RCPB bolts and therefore the radiation intensity should be less. But because of the large number of bolts associated the component supports, including pipe supports and restraints, the exposure values calculated for the RCPB bolts should provide a good indication of the exposures that will be incurred as a result of the proposed action of this VIA. Since occupational exposure will be a factor for only those plants that are operating (67), the exposures for this VIA are 67/47 times the PNL values. These values are 44 man-rem for the high estimate, 10 man-rem for the best estimate, and 3 man-rem for the low estimate.

6.4 PUBLIC PROPERTY

The effect of the proposed action on reducing the risk to public (offsite) property is calculated by multiplying the change in accident probability by a generic offsite property damage estimate. A mean value for the latter parameter is used that was determined by PNL for the VIA prepared for replacing the RCPB bolts⁴. "This estimate was derived from the mean value of results of CRAC2 calculations, assuming an SST1 release (major accident) for 154 reactors¹³." The following discounting formula, from NUREG/CR-3588¹², is employed to relate the costs to present value:

$$D = V[e^{-rt_i} - e^{-rt_f}]/r$$

where D = discounted value

V = damage estimate

t_i = years before reactor begins operation; 0 for operating reactors

t_f = years remaining until end of life

r = real discount rate

The damage estimate was converted to a present value by discounting it at a 10% rate¹⁴. For operating reactors with an average remaining lifetime of 22.2 years¹⁴, $D/V = 8.91$. The average remaining life time for reactors under construction is 27.1¹⁴ years and $D/V = 5.1$ if it is assumed that the average time before reactors under construction begin operation is 5 years. These values must be multiplied by the damage estimate, the number of affected PWRs (67 operating, 13 under construction¹⁴), and the change in core-melt frequency

to provide the total effect of this action on public property. It was indicated that the high and lower bounds are values for Indian Point 2 and Palo Verde 3 calculated from NUREG/CR-2723¹³. The results are shown in Table 6.11.

Table 6.11 Public Property Damage

	Offsite Property Damage (\$/event)	Discounted Offsite Property Damage \$		Discounted Offsite Property Damage [OP + CO] \$
		OP	CO	
Best Estimate	1.7E+09	3.2E+07	6.3E+06	3.8E+07
High Estimate	9.2E+09	1.6E+09	1.8E+08	1.8E+09
Low Estimate	8.3E+08	0	0	0

OP = Operating Plant
CO = Under construction

6.5 ONSITE PROPERTY

The effect of the proposed action on reducing the risk to onsite property is estimated by the following formula taken from NUREG/CR-3568¹²:

$$V_{OP} = N \times \Delta F \times U$$

where

V_{OP} = value of avoided onsite property damage
 N = number of affected facilities
 ΔF = change in accident frequency
 U = present value of property damage occurring with frequency F .

Suggested values¹² for U (present value of onsite property damage, conditional upon release; i.e., must be multiplied by accident frequency) are Best estimate = \$1.0E+10, High estimate = \$3.0E+10, and Low estimate = \$7.0E+09. The change in accident frequency (core melt) is 2.9E-04 for the high estimate and 3.2E-05 for the best estimate; and the number of operating PWRs is 67¹⁴. The results are:

Best Estimate	\$2.1E+07
High Estimate	\$5.8E+08
Low Estimate	0

6.6 INDUSTRY COSTS

Industry costs are a combination of the cost to implement improvements to their quality assurance programs (implementation costs) and the cost to perform the surveys, testing, and bolt replacements (operation costs). Implementation costs are a continuing cost since the quality programs not only have to be initially improved but also maintained in the improved state. Operation costs include labor, material, and equipment necessary to actually perform the proposed action that will decrease the likelihood of public risk caused by bolt failures.

6.6.1 Industry Implementation Costs

An important aspect of testing and then replacing faulty bolts with materials more resistant to corrosion is to assure that the material does indeed meet required specifications. The benefits sought from replacing existing bolts with those made of a different material would be degraded if the bolts received from the manufacturer did not meet specified standards. The VIA performed by PNL in 1985² stated "There are currently no definitive rules for the receiving inspection of bolts at nuclear sites¹⁵. The only requirements are general in nature. e.g., 10 CFR 50 APP B - Section VII, Control of Purchased Material, Equipment, and Services. It is basically up to the individual utility to develop a receiving inspection program for its purchased material." The present situation appears to be basically unchanged.

This analysis assumes that an upgraded receiving inspection for bolting materials will be implemented as part of the proposed action. An SRP for reactor bolting currently in revision offers guidelines on inspections to verify compliance with design requirements⁷. "Such inspection should include but be not limited to 1) surface nondestructive inspection (magnetic particle or dye penetrant) to preclude the possibility of preexisting flaws, and 2) mechanical testing to verify processing control. Such mechanical testing would include nondestructive hardness testing and destructive mechanical tests of lot samples."

The cost for implementation has been estimated by NRC Staff personnel as \$5000/yr., $\pm$ a factor of 2, for each plant. The cost is the same for a multiple plant site if the site utilizes identical units. A review of PWRs¹⁴ shows that 80 plants are operating or under construction (67 operating, 13 under construction). When identical plants at a single site are considered, there are 52 plant/sites with an average remaining life of 23.0 years. Remaining life is the difference between April 30, 1988 and 40 years after the construction permit was issued, the licensed life of the plant. This results in an industry implementation cost of $\$5.98\text{E}+06$ ($\$5000/\text{pl-yr} \times 52 \text{ pl} \times 23 \text{ yr}$) for the best estimate, $\$1.12\text{E}+07$ for the high estimate, and $\$2.99\text{E}+06$ for the low estimate. These estimates appear to be reasonable when compared with the high estimate of $\$1.5\text{E}+06$ obtained by PNL when performing the VIA for replacing the reactor coolant pressure boundary bolts².

6.6.2 Industry Operation Costs

This analysis assumes that industry operation costs due to this proposed action will consist of the cost for conducting a survey of the design and records for the bolts, performing tests on the installed bolts, and replacing those bolts that are faulty. The costs include labor, equipment that may need to be purchased, and the cost of new bolts. It is assumed that no extension of outage time will occur and, thus, no replacement power costs will result. Estimates for the industry operation costs have been provided by NRC Staff personnel and shown in Table 6.12. The costs are the same for a multiple plant site if the site utilizes identical units.

Table 6.12 Industry Operation Costs Per Plant

<u>Operation</u>	<u>Cost (\$)</u>	
	<u>Low</u>	<u>High</u>
Conduct Survey	2.0E+04	5.0E+04
Perform Tests	2.0E+05	3.0E+05
Replace Bolts	<u>1.0E+05</u>	<u>1.5E+05</u>
Total	3.2E+05	5.0E+05

Considering that there are 52 plant/sites¹⁴, the total industry operation costs are \$1.6E+07 for the low estimate and \$2.6E+07 for the high estimate as shown in Table 6.13. The best estimate has been calculated as the average of the high and low estimates and is \$2.1E+07.

Table 6.13 Total Industry Operation Costs

<u>Operation</u>	<u>Cost (\$)</u>		
	<u>Best Estimate</u>	<u>Low Estimate</u>	<u>High Estimate</u>
Conduct Survey	1.8E+06	1.0E+06	2.6E+06
Perform Tests	1.3E+07	1.0E+07	1.6E+07
Replace Bolts	<u>6.5E+06</u>	<u>5.2E+06</u>	<u>7.8E+06</u>
Total	2.1E+07	1.6E+07	2.6E+07

The estimated costs of Table 6.12 compare favorably with cost estimates that have been prepared for other value-impact analyses. A cost estimate was prepared for a sequence of steps to review, analyze, and repair or replace supports for steam generators and primary coolant pumps as a result of resolution of generic issue A-12, "Potential for Low Fracture

Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports" and reported in NUREG-0577¹⁰. A cost of \$45,000 was estimated to review specifications and available test data for materials that were installed in the supports, and then to perform some analysis of materials that were questionable. This cost compares well with the estimate of \$20,000 to \$50,000 for conducting the survey that is shown in Table 6.12.

PNL estimated the cost to replace only RCPB bolts in 47 PWRs as \$1.86E+06 to \$2.75E+06. The estimate in Table 6.13 of \$5.2E+06 to \$7.8E+06 to replace bolts in 52 plant sites involving 80 nuclear plants is reasonable considering that bolting for several systems other than the primary coolant system and for more nuclear plants is involved.

Since it is expected that testing will have to be performed on many more bolts than will be replaced and testing will likely involve removal and inspection for many bolts, it is expected that the testing of bolts will be more costly than replacement. Therefore the estimated cost of \$1.0E+07 to \$1.6E+07 to test as indicated in Table 6.13 is reasonable.

6.7 NRC COSTS

NRC staff has estimated that 0.1 to 1.0 man-yr will be required to issue requirements to licensees, review and approve licensee responses, and perform any required inspections. This estimate is based on the NRC costs that are being incurred in support of NRC Compliance Bulletin NO. 87-02, "Fastener Testing to Determine Conformance With Applicable Material Specifications"¹⁶. This bulletin requests "... that licensees 1) review their receipt inspection requirements and internal controls for fasteners and 2) independently determine, through testing, whether fasteners (studs, bolts, cap screws and nuts) in stores at their facilities meet required mechanical and chemical specification requirements." NRC costs, assuming \$120,000 per man-yr., are estimated to be \$12,000 for the low estimate, \$120,000 for the high estimate, and the best estimate being the average or \$66,000. NRC implementation costs to support the VIA for the RCPB bolts² was estimated to be \$9,700 to \$87,000, which compares favorably with the estimate provided by NRC staff for this VIA.

7.0 CONCLUSIONS

Bounding calculations, estimates of averted public risk, and costs to implement actions to decrease the potential for systems to fail as a result of bolt failures have been performed. In addition, estimates have been made for the averted occupational exposures, public property damage, and on site property damage.

Results of bounding calculations show that failure of bolts related to the emergency diesel generators and their supporting systems, such as the service water system; and component support bolts for the primary coolant system have the greatest potential to cause problems. The benefits of decreasing the potential for bolt failures in the other plant systems are approximately 5% of the benefits related to the primary system or the emergency diesel generators and their support systems. The emergency

diesel generators and their supporting systems are similar for most reactor types. Therefore, the impact of bolt failures in the emergency diesel generators and their supporting systems in PWRs are also applicable to other reactor types.

The best estimate of the value-impact ratio is 3708 \$/man-rem. The best estimate for the averted public risk is $7.3\text{E}+03$ man-rem and the best estimate of the impacts is $\$2.7\text{E}+07$ (industry and NRC costs). High estimates are $1.4\text{E}+05$ man-rem and $\$3.7\text{E}+07$ resulting in a value-impact ratio of 265 \$/man-rem.

Other decision factors such as occupational exposure, public property damage, and onsite property damage have also been considered. The occupational exposures are much smaller than the averted public risk (1010 man-rem best estimate and 9102 man-rem high estimate). The impacts resulting from public property and onsite property are much higher than those resulting from industry and NRC costs ($\$5.9\text{E}+07$ best estimate and $\$2.4\text{E}+09$ high estimate).

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APPENDIX A

TABLES 13.1-3. DEFINITION OF RELEASE CATEGORIES
USED IN SSPSA RISK MODEL

Release Category	Definition
S1	Early containment failure due to steam explosion or hydrogen burn with containment building sprays working.
$\overline{S1}$	Same as S1, but with containment building spray not working.
S2	Early containment leakage with late overpressurization failure and containment building sprays working.
$\overline{S2}$	Same as S2, but with containment building sprays not working.
$\overline{S2V}$	Same as S2, but with an additional vaporization component of the source term.
S3	Late overpressurization failure of the containment with no early leakage and containment building sprays working.
$\overline{S3}$	Same as S3, but with containment building sprays not working.
$\overline{S3V}$	Same as $\overline{S3}$, but with an additional vaporization component of the source term.
$\overline{S4V}$	Containment basemat melt-through with containment building sprays not working and additional vaporization component of the source term.
S5	Containment intact/isolated with enclosure air handling filtration working.
$\widehat{S5}$	Same as S5 but with enclosure air handling filtration not working.
S6	Containment bypass or isolation failure with containment building sprays working.
$\overline{S6V}$	Same as S6, but with containment building sprays not working and an additional vaporization component of the source term.

TABLE 12.5-85. MEAN EFFECTS CONDITIONAL ON RELEASE

MEAN EFFECTS WITH UNCERTAINTY USING SEABROOK MEAN FREQUENCIES

MEAN EFFECTS PER OCCURRENCE BY RELEASE CATEGORY AND EFFECT

EFFECT NAME	S1 USING S1	S2 USING S2 BAR V	S3 USING S3 BAR
ACUTE FATALITIES	9.983E+02	1.006E-01	0.000E+00
ACUTE INJURIES	1.654E+03	5.140E-01	8.868E-05
THYROID CANCERS	1.735E+03	3.746E+02	6.265E+01
LATENT CANCER FATALITIES	4.805E+03	1.052E+03	1.476E+02
MAN-REM	8.566E+06	5.368E+06	7.382E+05
PROPERTY DAMAGE	2.523E+08	1.318E+08	2.275E+07
INTERDICTION/DECONTAMINATED AREA, M**2	4.503E+07	6.807E+06	5.163E+05

EFFECT NAME	S5 USING S5	S6 USING S2 BAR V	S1 BAR USING S1 BAR
ACUTE FATALITIES	0.000E+00	1.006E-01	2.632E+02
ACUTE INJURIES	0.000E+00	5.140E-01	4.454E+02
THYROID CANCERS	3.143E-01	3.746E+02	4.662E+03
LATENT CANCER FATALITIES	4.268E-01	1.052E+03	6.743E+03
MAN-REM	2.142E+03	5.368E+06	2.572E+07
PROPERTY DAMAGE	2.000E+05	1.318E+08	1.141E+09
INTERDICTION/DECONTAMINATED AREA, M**2	0.000E+00	6.807E+06	3.378E+08

EFFECT NAME	S2 BAR USING S2 BAR V	S3 BAR USING S3 BAR	S5 MAT USING S5
ACUTE FATALITIES	1.006E-01	0.000E+00	0.000E+00
ACUTE INJURIES	5.140E-01	8.868E-05	0.000E+00
THYROID CANCERS	3.746E+02	6.265E+01	3.143E-01
LATENT CANCER FATALITIES	1.052E+03	1.476E+02	4.268E-01
MAN-REM	5.368E+06	7.382E+05	2.142E+03
PROPERTY DAMAGE	1.318E+08	2.275E+07	2.000E+05
INTERDICTION/DECONTAMINATED AREA, M**2	6.807E+06	5.163E+05	0.000E+00

TABLE 12.5-85 (continued)

MEAN EFFECTS WITH UNCERTAINTY USING SEABROOK MEAN FREQUENCIES

MEAN EFFECTS PER OCCURANCE BY RELEASE CATEGORY AND EFFECT

EFFECT NAME	S2 BAR V	S3 BAR V	S4 BAR V
	USING	USING	USING
	S2 BAR V	S3 BAR V	S3 BAR
ACUTE FATALITIES	1.006E-01	0.000E+00	0.000E+00
ACUTE INJURIES	5.140E-01	6.070E-04	8.868E-05
THYROID CANCERS	3.744E+02	4.078E+01	6.285E+01
LATENT CANCER FATALITIES	1.052E+03	1.036E+02	1.476E+02
MAN-REM	5.368E+06	5.097E+05	7.382E+05
PROPERTY DAMAGE	1.318E+08	1.945E+07	2.275E+07
INTERDICTED/DECONTAMINATED AREA, M**2	6.807E+06	2.462E+05	5.183E+05

EFFECT NAME	S6 BAR V
	USING
	S6 BAR V
ACUTE FATALITIES	1.751E+02
ACUTE INJURIES	2.211E+02
THYROID CANCERS	1.824E+03
LATENT CANCER FATALITIES	2.700E+03
MAN-REM	1.370E+07
PROPERTY DAMAGE	4.845E+08
INTERDICTED/DECONTAMINATED AREA, M**2	6.836E+07

TABLE 13.2-5a. RESULTS OF MATRIX OPERATION ϕ_{DMC}^1 ; FREQUENCY OF OCCURRENCE OF EACH RELEASE CATEGORY AS A RESULT OF EACH INITIATING EVENT

Initiating Events*		Release Categories*											
51	52	53	55	56	510000	520000	530000	550000	560000	570000	580000	590000	TOTAL
ELUCA	9-0110E-12	3-7102E-11	2-7457E-09	2-6274E-07	7-3408E-15	3-1372E-15	9-0155E-14	1-6435E-10	1-3257E-12	1-3257E-12	1-3257E-12	1-3257E-12	1-3257E-12
LLUCA	3-0777E-11	1-5601E-10	1-7157E-08	1-1222E-11	9-0155E-14	1-6435E-10	1-3257E-12	1-3257E-12	1-3257E-12	1-3257E-12	1-3257E-12	1-3257E-12	1-3257E-12
MLUCA	2-1495E-11	9-7107E-11	7-8521E-09	6-7022E-07	1-2413E-11	7-5708E-12	1-9837E-11	1-8021E-07	5-6947E-07	5-6947E-07	5-6947E-07	5-6947E-07	5-6947E-07
SLUCA	5-3596E-11	2-2475E-09	9-1161E-08	1-7202E-05	2-8135E-12	2-9655E-12	2-8086E-11	2-3357E-07	3-9851E-12	3-9851E-12	3-9851E-12	3-9851E-12	3-9851E-12
0	0	0	0	0	0	0	0	0	0	0	0	0	0
SCIA	9-0481E-12	8-3638E-10	5-5157E-09	6-7768E-07	5-3729E-13	2-2775E-12	5-2735E-12	1-3174E-07	3-2574E-12	3-2574E-12	3-2574E-12	3-2574E-12	3-2574E-12
AI	1-9748E-11	6-0777E-10	2-6295E-08	6-1646E-05	2-1019E-11	8-0679E-12	5-2735E-12	1-3174E-07	3-2574E-12	3-2574E-12	3-2574E-12	3-2574E-12	3-2574E-12
FI	1-2117E-11	5-0028E-10	3-9004E-08	3-6605E-08	1-6765E-11	7-7036E-12	1-2275E-10	3-9211E-08	1-9500E-11	1-9500E-11	1-9500E-11	1-9500E-11	1-9500E-11
LMF	1-2664E-11	5-3110E-11	2-2878E-09	9-0651E-07	7-7999E-14	2-0068E-12	2-2774E-12	1-4841E-08	2-0427E-12	2-0427E-12	2-0427E-12	2-0427E-12	2-0427E-12
PLMF	9-0798E-12	3-1889E-10	1-8078E-08	2-5737E-06	6-0745E-11	6-0128E-12	1-5541E-10	5-7735E-05	1-3101E-11	1-3101E-11	1-3101E-11	1-3101E-11	1-3101E-11
ELMF	6-8696E-12	3-5016E-10	2-7532E-08	2-7818E-06	6-0745E-11	3-6087E-12	8-0694E-12	1-7768E-05	1-3683E-11	1-3683E-11	1-3683E-11	1-3683E-11	1-3683E-11
LCF	2-6182E-12	1-1598E-10	6-1208E-08	8-0539E-07	1-8410E-11	6-5715E-13	6-5715E-13	8-8270E-05	3-9171E-12	3-9171E-12	3-9171E-12	3-9171E-12	3-9171E-12
SLIV	1-5120E-11	6-3431E-10	2-8748E-08	6-7493E-07	2-8493E-12	1-6526E-13	2-6073E-12	6-8952E-08	2-8911E-11	2-8911E-11	2-8911E-11	2-8911E-11	2-8911E-11
ANIV	1-2504E-13	5-2431E-12	2-1845E-10	5-0132E-08	7-7496E-16	3-6064E-11	7-3857E-13	2-1800E-08	2-7027E-13	2-7027E-13	2-7027E-13	2-7027E-13	2-7027E-13
CPFC	1-8788E-12	7-0049E-12	5-9465E-10	1-3736E-08	1-7690E-11	5-6608E-11	5-6608E-11	2-7652E-07	2-7652E-07	2-7652E-07	2-7652E-07	2-7652E-07	2-7652E-07
LDFF	3-9348E-12	1-5551E-10	1-1177E-08	1-0031E-05	3-5248E-12	1-3165E-12	7-7393E-11	1-2039E-06	5-5605E-12	5-5605E-12	5-5605E-12	5-5605E-12	5-5605E-12
SLI	2-1634E-11	1-0720E-10	3-9125E-08	6-8255E-06	3-5248E-12	1-3165E-12	7-7393E-11	1-2039E-06	5-5605E-12	5-5605E-12	5-5605E-12	5-5605E-12	5-5605E-12
SLRO	5-8422E-13	2-4208E-11	1-2810E-09	1-8504E-07	6-1736E-16	1-2880E-13	2-5364E-12	8-0022E-08	3-0115E-13	3-0115E-13	3-0115E-13	3-0115E-13	3-0115E-13
SLRV	5-8696E-12	1-4262E-11	1-2742E-09	1-8504E-07	6-1736E-16	1-2880E-13	2-5364E-12	8-0022E-08	3-0115E-13	3-0115E-13	3-0115E-13	3-0115E-13	3-0115E-13
SI	4-1354E-13	1-7379E-11	1-3287E-09	1-7253E-07	1-1724E-11	1-2822E-12	2-0152E-11	6-9900E-07	6-7627E-13	6-7627E-13	6-7627E-13	6-7627E-13	6-7627E-13
SLC	4-1354E-13	1-7379E-11	1-3287E-09	1-7253E-07	1-1724E-11	1-2822E-12	2-0152E-11	6-9900E-07	6-7627E-13	6-7627E-13	6-7627E-13	6-7627E-13	6-7627E-13
LDIC	2-2188E-12	2-0180E-10	5-3686E-08	1-3174E-06	1-1219E-11	1-2969E-12	1-5873E-11	5-8793E-12	1-4801E-07	8-6774E-12	8-6774E-12	8-6774E-12	8-6774E-12
LD5A	2-2177E-12	2-2168E-10	5-3686E-08	6-0875E-08	6-5358E-12	9-7731E-12	7-8004E-11	2-5160E-08	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0
ELPC	1-7294E-12	7-2716E-12	5-0645E-10	5-0940E-08	7-7269E-15	9-4025E-12	1-2017E-10	1-1387E-08	0	0	0	0	0
EL	2-8778E-11	1-1845E-10	9-2237E-11	8-7701E-08	3-7701E-16	3-9011E-12	3-5789E-12	2-0095E-08	5-2277E-13	5-2277E-13	5-2277E-13	5-2277E-13	5-2277E-13
ELOL	9-7824E-13	3-1059E-11	4-7690E-09	3-1037E-07	2-0716E-16	3-9011E-12	3-5789E-12	2-0095E-08	5-2277E-13	5-2277E-13	5-2277E-13	5-2277E-13	5-2277E-13
ELI	1-7505E-12	2-0356E-11	9-2457E-09	5-1131E-07	3-0084E-16	4-8415E-12	3-7157E-08	2-7127E-07	3-1935E-10	3-1935E-10	3-1935E-10	3-1935E-10	3-1935E-10
ELI	2-1370E-12	8-9131E-11	4-1958E-09	6-3755E-07	2-7475E-16	1-1545E-12	2-7996E-08	2-0050E-07	3-1935E-10	3-1935E-10	3-1935E-10	3-1935E-10	3-1935E-10
ELI	1-8407E-12	7-0595E-11	4-0466E-09	5-1808E-07	4-7406E-16	1-1510E-12	6-2994E-08	7-2274E-07	2-7019E-12	2-7019E-12	2-7019E-12	2-7019E-12	2-7019E-12
ELI	1-3224E-12	6-5995E-11	5-3596E-09	5-0358E-07	1-2214E-13	2-2777E-12	1-5435E-07	8-7721E-07	2-5708E-12	2-5708E-12	2-5708E-12	2-5708E-12	2-5708E-12
ELI	1-3901E-14	1-8426E-12	1-4260E-10	1-0575E-08	5-3686E-13	1-9691E-13	1-4031E-07	3-7022E-08	7-0038E-11	7-0038E-11	7-0038E-11	7-0038E-11	7-0038E-11
FSAC	0	0	0	0	0	0	0	0	0	0	0	0	0
FSAC	0	0	0	0	0	0	0	0	0	0	0	0	0
FSAC	0	0	0	0	0	0	0	0	0	0	0	0	0
FSAC	0	0	0	0	0	0	0	0	0	0	0	0	0
FC5A	0	0	0	0	0	0	0	0	0	0	0	0	0
FC5A	0	0	0	0	0	0	0	0	0	0	0	0	0
FC5A	0	0	0	0	0	0	0	0	0	0	0	0	0
FC5A	0	0	0	0	0	0	0	0	0	0	0	0	0
FE1	2-1607E-14	9-0641E-13	5-1862E-11	6-928	09	1-8130E-15	6-7635E-13	1-8841E-11	5-9980E-07	3-5814E-14	0	0	0
FE1	1-0804E-14	5-5121E-13	2-6910E-11	3-4642	09	9-6640E-15	3-3832E-13	1-3059E-12	2-9990E-07	1-7408E-14	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0	0	0	0	0	0
FPCC	0	0	0	0	0	0	0	0					

*Initiating event codes and release categories are defined in Tables 13.1-1 and 13.1-3, respectively.

TABLE 13.2-5b. RESULTS OF MATRIX OPERATION Φ_{DMC} (TRANSPPOSED); FREQUENCY OF OCCURRENCE OF ACCIDENT SEQUENCES GROUPED BY INITIATING EVENT FOR EACH PLANT DAMAGE STATE

Release Categories*	Initiating Event*												Sheet 1 of 3
	ELUCA	LLUCA	MLOCA	SLUCA	V	SGTR	KT	TT	TLNFM	PLMFM			
S1	9.0110E-12	3.8777E-11	2.1942E-11	5.3596E-11	0.	4.6415E-12	1.9248E-11	1.2113E-11	1.2654E-12	8.0798E-12			
S2	3.7102E-11	1.5963E-10	9.7107E-11	2.2475E-09	0.	8.3639E-10	8.0772E-10	5.0828E-10	5.3110E-11	3.3885E-10			
S3	2.7457E-09	1.2143E-08	7.8521E-09	9.1161E-08	0.	5.515E-09	6.2444E-08	3.9086E-08	2.2878E-09	1.4801E-08			
S5	2.6229E-07	1.1282E-06	6.9022E-07	1.7202E-05	0.	6.5784E-07	6.1654E-06	3.8055E-06	4.0641E-07	2.5924E-06			
S6	7.0490E-15	5.0344E-14	1.0413E-13	2.8134E-12	0.	5.5724E-13	2.1034E-11	1.2675E-11	3.7999E-14	2.9044E-13			
S1(BAK)	3.3372E-14	1.2272E-11	8.7578E-12	2.4655E-12	0.	2.2775E-12	8.0679E-12	4.7036E-12	2.0058E-12	6.0129E-12			
S2(BAK)	4.0143E-14	1.9859E-11	2.4778E-11	7.8986E-11	0.	5.4735E-12	1.9715E-10	1.2257E-10	2.2274E-12	1.5491E-10			
S3(BAK)	3.6435E-10	1.8023E-07	2.3357E-07	2.5066E-06	0.	1.3174E-07	6.3095E-06	3.7211E-06	6.8841E-08	4.9735E-06			
S5(HAT)	1.3247E-12	5.6982E-12	3.4845E-12	8.5400E-11	0.	3.2576E-12	3.0987E-11	1.9500E-11	2.0423E-12	1.3030E-11			
S2(VBAR)	3.2533E-14	8.9135E-12	4.5417E-12	9.9480E-12	0.	7.1706E-11	4.9194E-10	2.9478E-10	1.1977E-10	3.8376E-10			
S3(VBAR)	2.3461E-10	6.4189E-08	5.2973E-08	8.4837E-08	0.	8.2167E-07	5.0143E-07	1.7355E-07	8.5932E-07	2.4091E-07			
S4(VBAR)	1.5644E-10	4.2810E-08	6.5700E-09	1.0616E-08	0.	1.0156E-07	6.2302E-08	2.1654E-08	1.0621E-07	3.0037E-08			
S6(VBAR)	6.8923E-16	5.2592E-13	2.8537E-13	1.0161E-11	1.8400E-06	9.1023E-09	2.5054E-11	1.4939E-11	3.9284E-14	3.0011E-13			
TOTAL	2.6584E-07	1.4277E-06	9.7134E-07	1.9897E-05	1.5400E-06	1.7183E-06	1.3104E-05	8.0368E-06	1.4433E-06	7.8530E-06			
S1	8.4869E-12	2.7618E-12	1.5126E-11	1.2504E-13	1.6788E-13	3.4438E-12	2.1634E-11	5.8421E-13	5.0969E-13	4.1344E-13			
S2	3.5614E-10	1.1585E-10	6.3439E-10	5.2434E-12	7.0449E-12	1.4451E-10	9.0725E-10	2.4206E-11	1.8924E-11	1.7349E-11			
S3	2.7532E-08	6.1208E-09	2.8784E-08	2.1844E-13	5.4464E-10	1.1172E-08	3.9142E-08	1.2810E-09	1.2427E-09	1.2427E-09			
S5	2.7189E-06	8.8589E-07	4.8537E-06	4.0132E-06	5.3784E-08	1.1033E-06	6.9425E-06	1.8509E-07	1.4444E-07	1.3245E-07			
S6	8.9424E-12	1.2410E-13	2.2939E-12	7.2494E-16	1.7690E-13	3.6288E-12	2.7482E-13	6.1734E-14	8.2874E-14	4.1728E-13			
S1(BAK)	3.3487E-12	3.4715E-13	1.6426E-13	4.0364E-14	5.4608E-13	3.7599E-11	3.9369E-13	1.2344E-13	8.2860E-13	1.2822E-12			
S2(BAK)	8.6494E-11	3.1655E-12	2.8075E-12	7.3857E-13	5.8064E-12	3.7665E-12	7.6280E-12	2.5364E-12	1.5521E-11	2.0152E-11			
S3(BAK)	2.7680E-06	8.8220E-08	8.4452E-08	2.3800E-08	2.5656E-08	1.2039E-06	2.4486E-07	8.0022E-08	4.9983E-07	6.4896E-07			
S5(HAT)	1.3663E-11	4.5175E-12	2.4391E-11	2.0167E-13	2.7027E-13	5.5400E-12	3.4887E-11	9.3011E-13	7.2580E-13	6.6560E-13			
S2(VBAR)	2.0465E-10	6.9024E-11	5.3514E-11	8.2400E-13	1.2993E-11	8.5075E-11	4.7252E-12	2.1968E-12	1.7551E-11	3.0507E-11			
S3(VBAR)	1.3510E-07	1.0643E-07	3.1506E-08	6.3107E-09	1.0764E-07	5.4824E-08	5.5134E-08	1.5751E-08	1.2398E-07	2.5290E-07			
S4(VBAR)	1.6842E-08	1.3155E-08	3.8984E-09	7.8121E-13	1.3318E-08	6.8386E-09	6.8270E-09	1.9509E-09	1.5350E-08	3.1291E-08			
S6(VBAR)	1.0572E-11	3.4648E-15	2.7120E-12	2.0225E-17	2.0914E-13	4.2901E-12	2.8987E-13	3.5758E-12	1.5907E-11	4.9095E-13			
TOTAL	5.6671E-06	1.1100E-06	5.0036E-06	7.1249E-08	4.5183E-07	2.3803E-06	7.2894E-06	2.8413E-07	7.8491E-07	1.0670E-06			
S1	4.8268E-12	5.2778E-12	0.	0.	1.7529E-12	2.8776E-13	9.7626E-13	1.6750E-12	2.1370E-12	1.6807E-12			
S2	2.0389E-10	2.2162E-10	0.	0.	7.2150E-12	1.1845E-12	4.1058E-11	7.0356E-11	8.9713E-11	7.0545E-11			
S3	8.9365E-08	2.5689E-08	0.	0.	5.5064E-10	9.2237E-11	8.3090E-09	9.1458E-09	9.1458E-09	6.4046E-09			
S5	1.5179E-06	1.6875E-06	0.	0.	5.0989E-08	8.3701E-09	3.1077E-07	5.3513E-07	6.8375E-07	5.3808E-07			
S6	1.1219E-11	6.6361E-12	0.	0.	4.4288E-15	7.5182E-16	2.0736E-14	3.8089E-14	7.4735E-14	8.2408E-14			
S1(BAK)	1.2986E-10	9.9144E-13	2.7733E-12	1.5305E-12	9.2055E-12	3.9301E-14	6.9631E-14	4.8216E-13	1.1545E-12	1.4340E-12			
S2(BAK)	2.1460E-10	5.6735E-12	7.8044E-11	4.3091E-11	1.3207E-10	3.6996E-11	1.2298E-09	1.8492E-08	5.2949E-08	6.9244E-08			
S3(BAK)	4.9000E-06	1.8041E-07	2.5160E-06	1.3892E-06	7.1435E-08	2.0098E-08	6.1428E-08	3.2813E-07	7.0850E-07	7.2827E-07			
S5(HAT)	7.6273E-12	8.4799E-12	0.	0.	2.5752E-13	4.2273E-14	1.5817E-12	2.6801E-12	3.4359E-12	2.7039E-12			
S2(VBAR)	7.7649E-08	4.8055E-11	2.3845E-13	8.5883E-14	3.4401E-07	5.0840E-07	1.3752E-06	1.4952E-06	2.4265E-06	2.7124E-06			
S3(VBAR)	5.5389E-05	3.5287E-07	2.4319E-09	7.8600E-08	3.8524E-08	9.4069E-10	5.3927E-08	1.6690E-07	2.4160E-07	2.4160E-07			
S4(VBAR)	6.8461E-06	4.3622E-08	4.3141E-10	1.9785E-10	5.2407E-08	2.5684E-08	1.1946E-10	6.6822E-09	2.0685E-08	3.4843E-08			
S6(VBAR)	9.0024E-15	1.3183E-13	8.4605E-16	4.6667E-16	2.7300E-08	2.0746E-08	4.4200E-12	1.0835E-09	6.8613E-09	4.1621E-08			
TOTAL	6.8821E-05	2.2940E-06	2.5189E-06	1.3904E-06	6.2545E-07	6.2196E-07	1.7580E-06	2.4480E-06	4.0754E-06	4.4126E-06			

*Initiating events and release categories are defined in Tables 13.1-1 and 13.1-3, respectively.

TABLE 13.2-5b (continued)

Initiating Event*											Sheet 2 of 3	
E.71	E1.0T	FSKCC	FSKAC	FCRCC	FCRSW	FCRAC	FET1	FET3	FPCC			
S1	1.5724E-12	4.3901E-14	0.	0.	0.	0.	2.1607E-14	1.0804E-14	0.			
S2	6.5942E-11	1.8425E-12	0.	0.	0.	0.	9.0641E-13	4.5321E-13	0.			
S3	5.5356E-09	1.6263E-10	0.	0.	0.	0.	5.3821E-11	2.6910E-11	0.			
S5	5.0358E-07	1.4057E-08	0.	0.	0.	0.	6.3284E-09	3.4642E-09	0.			
S6	1.2214E-13	5.3699E-15	0.	0.	0.	0.	1.8136E-14	9.0680E-15	0.			
S11BAR1	2.5277E-12	1.9461E-13	3.9639E-12	1.0381E-12	9.9098E-12	4.2004E-12	6.7633E-13	3.3812E-13	4.6246E-12			
S21BAR1	1.5645E-07	1.4031E-08	1.1160E-10	0.	2.7901E-10	6.5037E-11	0.	1.8612E-11	9.3059E-12	1.3020E-10		
S31BAR1	8.7873E-07	3.7022E-06	3.5978E-06	0.	8.9446E-06	2.0966E-06	0.	5.9980E-07	2.9990E-07	4.1975E-06		
S51MAT1	2.5306E-12	7.0636E-14	0.	0.	0.	0.	0.	3.4816E-14	1.7408E-14	0.		
S2V1BAR1	6.7572E-06	1.2721E-06	2.2243E-13	6.4884E-11	5.5607E-13	1.3871E-13	2.6254E-10	9.9834E-13	4.9917E-13	2.5450E-13		
S3V1BAR1	6.9466E-07	6.8477E-08	2.6321E-09	4.6196E-07	6.5802E-09	2.0266E-09	1.8642E-06	7.4682E-09	3.7341E-09	3.0708E-09		
S4V1BAR1	8.5902E-08	8.4624E-09	5.1242E-10	5.7096E-08	1.2811E-09	3.5951E-10	2.3102E-07	9.5422E-10	4.7711E-10	5.9782E-10		
S6V1BAR1	1.8309E-07	3.2505E-08	1.2086E-15	2.5732E-18	3.3216E-15	7.0504E-16	1.0412E-17	4.1185E-15	2.0592E-15	1.4101E-15		
TOTAL	9.2652E-06	1.4468E-06	3.6011E-06	5.1912E-07	9.0028E-06	2.0991E-06	2.1005E-06	6.1523E-07	3.0761E-07	4.2013E-06		
S1	3.4540E-13	8.0281E-15	1.4212E-14	4.2535E-15	0.	4.7503E-15	0.	9.3924E-19	0.	0.		
S2	1.4494E-11	3.3264E-13	5.8506E-14	1.7838E-13	0.	1.9921E-13	0.	3.9405E-17	0.	0.		
S3	9.4068E-10	1.7602E-11	4.4506E-12	7.4304E-12	0.	8.2840E-12	0.	2.5541E-15	0.	0.		
S5	1.1074E-07	2.5435E-09	4.1350E-10	1.3651E-09	0.	1.5246E-09	0.	3.0109E-13	0.	0.		
S6	4.9960E-14	8.4833E-16	1.8451E-17	2.4643E-17	0.	2.5602E-17	0.	1.4824E-19	0.	0.		
S11BAR1	7.0861E-12	4.2905E-14	4.4979E-15	1.3730E-15	7.9608E-13	2.3609E-17	1.3984E-14	5.6700E-17	6.0120E-15	1.1601E-14		
S21BAR1	9.5135E-12	6.3951E-13	7.2782E-15	2.5123E-14	0.	5.8944E-16	3.9371E-13	3.4646E-17	1.6927E-13	0.		
S31BAR1	2.9787E-07	2.0603E-08	6.6054E-11	8.0958E-10	0.	1.8981E-11	1.2692E-08	1.0835E-12	5.4567E-09	0.		
S51MAT1	5.5650E-13	1.2781E-14	2.0884E-15	6.8000E-15	0.	7.6612E-15	0.	1.5130E-18	0.	0.		
S2V1BAR1	5.7780E-10	7.7095E-13	3.2668E-15	3.0016E-14	4.7757E-11	1.1114E-16	7.8468E-16	3.4674E-15	3.3735E-16	7.2510E-13		
S3V1BAR1	3.0075E-06	9.0075E-09	2.3525E-11	2.1467E-10	3.5426E-07	1.2141E-12	4.2854E-12	2.4701E-11	3.9920E-12	5.1625E-09		
S4V1BAR1	7.1713E-07	1.1143E-09	1.5690E-11	2.6574E-11	4.3784E-08	1.5104E-13	1.8077E-12	3.0530E-12	7.717E-13	6.3806E-10		
S6V1BAR1	4.0011E-17	4.9138E-14	1.9275E-16	6.8799E-19	1.9733E-18	4.9855E-19	4.2638E-18	1.4320E-22	1.8331E-18	2.8756E-20		
TOTAL	3.7894E-06	3.3287E-08	5.2331E-10	2.4236E-09	3.9809E-07	1.5534E-07	1.2704E-08	2.9145E-11	5.4617E-09	5.8013E-09		
APC	ACK	APAB	FLLP	FL1SC	FL2SC	FL3SC	FL5M	TC1L	ALUMF	ALOSP		
S1	1.0304E-10	0.	0.	1.8425E-13	3.4839E-13	0.	0.	1.5891E-13	3.9072E-12	1.2639E-14		
S2	0.	0.	0.	7.7299E-12	1.3779E-11	0.	0.	6.6671E-12	1.6386E-10	5.3006E-13		
S3	0.	0.	0.	5.0170E-10	1.0124E-09	0.	0.	4.3271E-10	6.8712E-09	2.4598E-11		
S5	0.	0.	0.	5.9063E-08	1.0522E-07	0.	0.	5.0942E-08	1.2540E-06	4.0553E-09		
S6	0.	0.	0.	2.6592E-14	4.8751E-14	0.	0.	2.2936E-14	2.6555E-14	3.8867E-16		
S11BAR1	0.	2.7803E-13	2.2022E-13	3.7799E-12	4.2043E-13	1.7005E-13	1.7608E-12	3.3584E-12	1.6776E-11	1.6415E-14		
S21BAR1	0.	0.	6.2002E-12	5.0739E-12	2.8816E-13	0.	4.9552E-11	4.3762E-12	4.5764E-13	1.5052E-14		
S31BAR1	0.	0.	1.9988E-07	9.1591E-09	0.	1.5974E-06	1.3702E-07	1.4745E-08	4.6503E-10	0.		
S51MAT1	0.	0.	0.	2.9680E-13	5.2876E-13	0.	0.	2.5599E-13	6.3013E-12	2.0378E-14		
S2V1BAR1	0.	1.7378E-11	1.2357E-14	3.0818E-10	2.5591E-11	1.0629E-11	1.5140E-13	2.6949E-10	6.2849E-10	9.9327E-13		
S3V1BAR1	0.	1.2372E-07	1.4623E-10	1.6043E-06	1.9261E-07	7.5673E-08	1.5440E-09	1.4274E-06	7.4580E-06	7.0767E-09		
S4V1BAR1	0.	1.5292E-08	2.8468E-11	1.9829E-07	2.2570E-08	9.3528E-09	2.7391E-10	1.7643E-07	9.2178E-07	8.7470E-10		
S6V1BAR1	0.	6.8916E-19	6.7147E-17	2.1339E-17	1.0630E-18	4.2152E-19	5.3717E-16	1.8403E-17	4.1756E-11	1.2989E-18		
TOTAL	1.0304E-10	1.3403E-07	2.0006E-07	2.0213E-06	3.2061E-07	8.5036E-08	1.5993E-06	1.7925E-06	9.6562E-06	1.2498E-08		

*Initiating events and release categories are used in Tables 13.1-1 and 13.1-3, respectively.

TABLE 13.2-5b (continued)

Release Categories*	ATF	ASLUC	E-2A	Initiating Event*					E1.0A
				E-3A	E-4A	E-5A	E-7A	E1.0A	
S1	6.3323E-12	4.9401E-15	1.4148E-15	4.4357E-15	2.6079E-16	1.0446E-13	2.6559E-13	2.0420E-14	
S2	2.6556E-10	2.0435E-13	5.4335E-14	1.4604E-14	1.0938E-14	4.3833E-12	1.1153E-11	8.5661E-13	
S3	1.1126E-08	1.3122E-11	2.6125E-12	4.0814E-13	5.3141E-13	3.2100E-10	7.3472E-10	5.0374E-11	
S5	2.0323E-06	1.5994E-09	4.5402E-10	1.4231E-10	8.3668E-11	3.3473E-08	8.5135E-08	6.5479E-09	
S6	1.1498E-12	5.7062E-16	1.1743E-17	3.4808E-18	1.4199E-15	2.9532E-15	6.5256E-15	5.0240E-16	
S11BAR)	5.3763E-13	5.7021E-15	8.3866E-17	3.0784E-12	2.7201E-13	4.8248E-13	1.3821E-12	3.5599E-13	
S21BAR)	8.3388E-13	1.5774E-13	9.9859E-15	2.1186E-14	8.7171E-14	4.3240E-11	1.7614E-10	2.2847E-11	
S31BAR)	2.6472E-08	5.0852E-09	1.6090E-12	1.3663E-12	4.6257E-10	4.0751E-09	1.6076E-08	2.0804E-09	
S51HAT)	1.0212E-11	8.0344E-15	2.2815E-15	7.1512E-16	4.2044E-16	1.6820E-13	4.2782E-13	3.2904E-14	
S2VIBAR)	1.9056E-11	4.0920E-15	7.4911E-12	1.2544E-10	3.6284E-11	5.8992E-08	5.4298E-07	3.6730E-07	
S3VIBAR)	2.2609E-07	4.7957E-11	3.6532E-11	1.3699E-06	2.5432E-07	2.1271E-07	6.0717E-07	1.5740E-07	
S4VIBAR)	2.7945E-08	6.1917E-12	4.5153E-12	1.6931E-07	3.1433E-08	2.6290E-08	7.5044E-08	1.4454E-08	
S6VIBAR)	1.2790E-12	2.3680E-15	4.1563E-12	1.5391E-07	2.8617E-08	2.0513E-08	5.2579E-08	1.5092E-08	
TOTAL	2.3246E-06	6.7527E-09	5.1101E-10	1.6934E-05	3.1495E-07	3.5442E-07	1.3749E-06	5.6795E-07	

*Initiating events and release categories are defined in Tables 13.1-1 and 13.1-3, respectively.

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11. ABSTRACT *(200 words or less)*

Generic Safety Issue (GSI)-29 deals with staff concerns about public risk due to degradation or failure of safety-related bolting in nuclear power plants.

The issue was initiated in November 1982. Value-impact studies of a mandatory program on safety-related bolting for operating plants were inconclusive; therefore, additional regulatory requirements for operating plants could not be justified in accordance with provisions of 10 CFR 50.109. In addition, based on (1) operating experience with bolting in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry-proposed actions, the staff concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic failure of a reactor coolant pressure boundary joint that will lead to significant accident sequences is highly unlikely. For future plants, it was concluded that a new Standard Review Plan section should be developed to codify existing bolting requirements and industry-developed initiatives.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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