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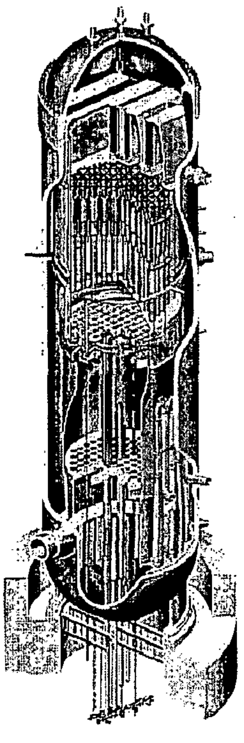
BWRVIP-06NP, Revision 1-A: BWR Vessel and Internals Project

Safety Assessment of BWR Reactor Internals



BWRVIP-06NP, Revision 1-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals

1019058NP

Final Report, March 2010

EPRI Project Manager
J. Hosler

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NRC SAFETY EVALUATION

In accordance with an NRC request, the NRC Safety Evaluation immediately follows this page. Other NRC and BWRVIP correspondence on this subject are included in appendices.

Note: The changes proposed by the NRC in this Safety Evaluation, as well as these proposed in response to NRC Requests for Additional Information, have been incorporated into the current version of the report (BWRVIP-06, Revision 1-A).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 29, 2008

Mr. Rick Libra
Exelon
Chairman, BWR Vessel and Internals Project
Electric Power Research Institute
3420 Hillview Avenue
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SUBJECT: SAFETY EVALUATION FOR ELECTRIC POWER RESEARCH INSTITUTE (EPRI) BOILING WATER REACTOR (BWR) VESSEL AND INTERNALS PROJECT (BWRVIP) TOPICAL REPORT (TR)-1006598, "BWRVIP-06-A: BWR VESSEL AND INTERNALS PROJECT, SAFETY ASSESSMENT OF BWR REACTOR INTERNALS, REVISED SECTION 4.0: CONSIDERATION OF LOOSE PARTS" (TAC NO. MC7448)

Dear Mr. Libra:

By letter dated May 24, 2002 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML021500624), the BWRVIP submitted "BWRVIP-06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. The purpose of the TR is to exchange information with the NRC staff in support of generic regulatory improvements related to assessing the safety consequences of potential cracking of BWR internals. The NRC staff concluded that no technical changes were necessary or required for the approval of the TR and by letter dated September 16, 2003 (ADAMS Accession No. ML032650767) informed the BWRVIP of their acceptance and approval of BWRVIP-06.

By letter dated May 11, 2005 (ADAMS Accession No. ML051370191), the BWRVIP submitted "BWRVIP-06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals, Revised Section 4.0: Consideration of Loose Parts," to the NRC staff for review. Only Section 4.0 of the previously NRC staff approved TR (BWRVIP-06-A) was revised. The purpose of the revised section of the TR is to provide a general evaluation of the potential impact of loose parts generated in the reactor vessel due to cracking of reactor vessel internal components and to assess how loose parts affect the safe shutdown of the reactor plant and offsite dose.

By letter dated December 21, 2006 (ADAMS Accession No. ML070030188), the NRC staff sent a request for additional information (RAI) to the BWRVIP. By letter dated August 9, 2007 (ADAMS Accession No. ML072250216), the BWRVIP stated that the response to the RAI would be sent to NRC by November 30, 2007. By letter dated November 30, 2007, the response to the RAI was received by the NRC staff (ADAMS Accession No. ML073410041).

The NRC staff has reviewed, evaluated, and determined that revised Section 4.0 of BWRVIP-06-A supports the generic regulatory improvements related to assessing the safety consequences of potential cracking of BWR internals. Further, the NRC staff finds that it is acceptable for referencing in licensing documentation for operating BWR/2-6 plants to the extent specified and under the limitations delineated in the TR and in the enclosed SE. The SE defines the basis for our acceptance of the revised Section 4.0 of BWRVIP-06-A.

Our acceptance applies only to material provided in the revised Section 4.0 of BWRVIP-06-A. We do not intend to repeat our review of the acceptable material described in the revised Section 4.0 of TR BWRVIP-06-A. When TR BWRVIP-06-A appears as a reference in licensing documentation, our review will ensure that the material presented applies to the specific plant involved. Licensees will be expected to implement the provisions of BWRVIP-06-A, including the revised Section 4.0, subject to the limitations identified in the enclosed SE, as part of their BWRVIP program unless deviations from the program are justified. Licensees shall identify such deviations to the NRC in accordance with BWRVIP program requirements.

In accordance with the guidance provided on the NRC website, we request that the BWRVIP publish accepted proprietary and non-proprietary versions of this BWRVIP report within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC staff RAIs and your responses. The accepted versions shall include an "-A" (designating accepted) following the BWRVIP report identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this BWRVIP report, the BWRVIP and/or licensees referencing it will be expected to revise the BWRVIP report appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Mark J. Maxin

Mark J. Maxin, Acting Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure: SE

cc w/ encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT

"BWRVIP-06-A: BWR [BOILING WATER REACTOR] VESSEL AND INTERNALS

PROJECT [(BWRVIP)], SAFETY ASSESSMENT OF BWR REACTOR INTERNALS,

REVISED SECTION 4.0: CONSIDERATION OF LOOSE PARTS"

BWRVIP

PROJECT NO. 704

1.0 INTRODUCTION AND BACKGROUND

By letter dated May 11, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051370191), the BWRVIP submitted "BWRVIP-06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals, Revised Section 4.0: Consideration of Loose Parts" to the U.S. Nuclear Regulatory Commission (NRC) staff for review. Only Section 4.0 of the previously NRC approved TR (BWRVIP-06-A) was revised in its entirety by the BWRVIP. The revision of Section 4.0 provides a general evaluation of the potential impact of loose parts generated in the reactor vessel due to cracking of reactor vessel internal components (CVIP) and assesses how these loose parts may affect the safe shutdown of the reactor plant and offsite dose. Operating experience of most plants indicates that loose parts have not significantly affected plant operations.

By letter dated December 21, 2006 (ADAMS Accession No. ML070030188), the NRC staff sent a request for additional information (RAI) to the BWRVIP. By letter dated August 9, 2007 (ADAMS Accession No. ML072250216), the BWRVIP stated that the response to the RAI would be sent to the NRC staff by November 30, 2007. By letter dated November 30, 2007 (ADAMS Accession No. ML073410041), the NRC staff received the response to the RAI. The purpose of this NRC staff safety evaluation (SE) is to determine the acceptability of the revised Section 4.0.

The original BWRVIP-06 TR was submitted to the NRC staff for review by letter dated May 24, 2002 (ADAMS Accession No. ML021500624). The purpose of the TR was to exchange information with the NRC staff in support of generic regulatory improvements related to assessing the safety consequences of potential cracking of BWR internals.

Increased occurrence of identified intergranular stress corrosion cracking (IGSCC) in BWR internals prompted the U.S. BWR executives to form the BWRVIP in June 1994, to address integrity issues arising from service-related degradation of these important components. It was apparent to the BWRVIP and to the NRC staff that as inspection techniques improve and as more inspections are performed, additional IGSCC related cracking in welded and bolted locations of reactor internals would be identified. On this basis, the BWRVIP submitted

ENCLOSURE

TR BWRVIP-06 to the NRC staff to exchange information and to support generic regulatory efforts related to assessing the safety consequences of potential cracking of BWR/2-6 reactor internals. In addition, TR BWRVIP-06 supports the determination of the short- and long-term actions required to ensure safe operation with the potential for component cracking. TR BWRVIP-06 generically evaluates postulated failures caused by IGSCC in welded and bolted locations of reactor vessel internals (RVIs) and establishes long-term actions which the BWRVIP stated are appropriate to ensure continued safe operation. The assessment considers RVIs functions during normal, transient, seismic, and design basis accident conditions. The results of TR BWRVIP-06 are intended to provide utilities with a generic reactor internals management plan which can be tailored to meet the needs of individual utilities. Additionally, TR BWRVIP-06 is intended to provide the NRC staff with information needed to evaluate future cracking in BWR internal components.

The NRC staff concluded that no technical changes were necessary or required for the approval of TR BWRVIP-06 and by letter dated September 16, 2003 (ADAMS Accession No. ML032650767), the NRC staff informed the BWRVIP of their acceptance and approval of TR BWRVIP-06.

The revised Section 4.0 of TR BWRVIP-06-A applies only to operating BWR/2-6 plants.

2.0 REGULATORY EVALUATION

Guidance applicable for loose parts can be found in the General Design Criteria (GDC) 1 and 13 of Appendix A of 10 CFR Part 50. GDC 1 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that a quality assurance program be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. GDC 13 requires, in part, that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences (AOOs), and for accident conditions to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the core, and the reactor coolant pressure boundary. Additional applicable guidance is provided in NRC Regulatory Guide (RG) 1.133, Revision 1, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

Revised Section 4.0 of BWRVIP-06-A provides a qualitative rationale for any radiation releases to be within acceptable plant-specific limits as specified in Part 100, "Reactor Site Criteria," of Title 10 of the *Code of Federal Regulations* (10 CFR). Revised Section 4.0 states that smaller loose parts may pass through a fuel lower tie plate, be trapped on a spacer, and may wear a hole in the fuel cladding. As a result of this fuel fretting, fuel cladding leakage may occur, which would be detected by the off-gas system so that appropriate actions could be taken to maintain the off-gas radiation release within acceptable 10 CFR Part 100 limits. All identified technical specification actions will be complied with as required. There are possible operational concerns that a smaller loose part could cause fuel fretting; however, such concerns do not constitute a safety issue with respect to safe shutdown and offsite dose.

In addition, the revised Section 4.0 describes the impact of loose parts on the recirculation system. Smaller loose parts are expected to pass through the recirculation pump without causing any damage or detectable flow reduction in jet pump drive flow. Large and heavy items in the past have passed through the pump and the jet pump nozzles without causing any damage. If a loose part prevents the recirculation discharge valve from fully closing, the loss of low pressure coolant injection (LPCI) flow will delay the core re-flooding leading to an increase in the peak cladding temperature (PCT). Revised Section 4.0 provides a rationale, based on a bounding evaluation using realistic models and assumptions with no credit for LPCI or discharge valve closure, that one core spray system is sufficient to keep the PCT below the 10 CFR 50.46 limit of 2200° F presenting no significant safety hazard.

The NRC staff reviewed the revised Section 4.0 of TR BWRVIP-06-A, and concluded that the safety assessment of potential impact of loose parts generated in a BWR, as provided, is consistent with the above criteria and guidance.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed "BWRVIP-06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals, Revised Section 4.0: Consideration of Loose Parts." The revised Section 4.0 presents a general and a qualitative type consideration of loose parts. It addresses safety concerns and operational concerns from postulated loose parts.

The evaluation addresses the potential impact of loose parts generated in the vessel due to cracking of RVIs. Additionally, evaluation is included to assess why loose parts do not negatively affect safe shutdown and offsite dose. This analysis is also valid for loose parts generated outside the reactor vessel, as long as they reach the regions inside the vessel considered as source locations, and their sizes are within the range of sizes considered in this evaluation. Operating experience of most plants indicates that loose parts have not significantly affected plant operation. However, there has been some degradation of certain components and/or systems at times. It is important to ensure that programs are in place to effectively eliminate the introduction of loose parts, promptly identify loose parts that enter the vessel, and implement appropriate corrective action upon identification of loose parts wherever they may be. The evaluation provided in the TR is general in nature and does not assess the effects on the basis of individual part geometries or material properties, as these are resolved on a case-by-case basis. The generation of multiple loose parts that could arrange themselves in a manner that would cause unacceptable conditions was not considered to be credible for the purpose of this evaluation.

Only loose parts that are not detectable and could impact the safe plant operation and shutdown capability are generally considered to be of safety significance. Loose parts that are detectable due to their observable collateral impact on plant operation would indicate that an abnormal plant condition exists and the plant would normally be brought to a safe condition by operator action.

Because several plant-specific features can determine the acceptable loose part size limits for safety and operational concerns, only general criteria were provided.

3.1 Loose Parts and Structural Issues

The safety concerns addressed in this SE are the potential of loose parts to interfere with the main steam isolation valves (MSIV), safety/relief valve (SRV) operation, control rod operation, high pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC), reactor water clean up (RWCU), or residual heat removal (RHR) isolation valves, nuclear instrumentation, and operation of RHR pumps and heat exchangers. Also addressed are the potential for fuel damage due to fuel bundle flow blockage, impact damage on reactor internals, and corrosion or adverse chemical reaction with other reactor materials. The operational concerns evaluated are potential for fuel fretting, interference with operation of RWCU pumps, heat exchangers and filter demineralizers, flow blockage of the reactor vessel bottom head drain, and impairment of recirculation system performance. The NRC staff finds that the TR has adequately addressed the safety and operational concerns due to loose parts from qualitative considerations.

The revised Section 4.0 considers transport of loose parts in the reactor vessel and associated systems from gravity, flow, or combined gravity and flow considerations. The possibility of loose parts being carried over by the flow to locations where they could present a safety concern or affect safe shutdown and offsite dose is evaluated. The evaluation also addresses the source location of loose parts that have been lost in or migrated to the upper plenum, steam separator/dryer, downcomer, and lower plenum regions. From geometric considerations, the loose parts can be in a variety of sizes and shapes classified into three categories, namely large parts greater than 2 inches in size, small parts 1 to 2 inches in size, and debris smaller than 1 inch in size. The NRC staff finds that the evaluation of loose parts from the source location, transport, size, and shape considerations to be acceptable.

The NRC staff sought information about large loose parts resulting from steam dryer failure becoming missiles that could damage other reactor components. For example, large loose parts entering the Reactor Recirculation System (RRS) could cause damage to recirculation pump. The BWRVIP indicated that normal monitoring activities of the recirculation system would ensure that either damage to the pump or recirculation system performance degradation in terms of flow reduction could be noted and appropriate steps taken to mitigate the condition. The NRC staff finds this acceptable.

The NRC staff questioned the BWRVIP on the potential for impact of loose parts on the standby liquid control (SLC) system standpipe. The BWRVIP responded that the differential pressure (DP) line and the liquid control line are supported at the vessel nozzle, the shroud support skirt, and along the shroud as it is routed inside the core. This results in a configuration that is a well supported pipe within a pipe configuration and is unlikely to be damaged from impact by a loose part entering the lower plenum. The injection of sodium pentaborate is not prevented even if the pipe attached to the core shroud, which is for core DP and not SLC, is damaged by a loose part. Also, failure of the pipe in the lower plenum, which results in a change of measured core DP, will be detected in the control room. The NRC staff finds the clarification provided by BWRVIP on the impact of loose parts on SLC standpipe is acceptable.

The NRC staff sought information regarding fuel bundle upper tie plate pass through sizes. The BWRVIP provided a table of maximum pass through sizes for rectangular, cylindrical, and square shaped loose parts for various General Electric (GE) nuclear fuel designs. The NRC staff sought a discussion on whether there is a potential for both MSIVs failing to close due to large loose parts from steam dryer failure. BWRs are designed with a redundant valve for valves in a line that require redundancy. From past experience, it is highly unlikely that two large loose parts from the steam dryer would migrate down a single main steam line and bend or break both of the legs on the vertical center line of both MSIVs. One potential loose part and definitely more than one from a failed steam dryer will be detected by changes in steam quality.

x

If a large part is lodged in a main steam line, control room indicators will detect the mismatch in steam flow. The above rationale provided by the BWRVIP, regarding the unlikelihood of both MSIVs failing due to large loose parts from steam dryer failure and the detection of the effects indicated by the changes in moisture carryover and the main steam flow imbalance, is acceptable to the NRC staff.

The BWRVIP addressed the NRC staff question on the possibility of loose parts drawn into the SRV standpipe and SRV valve due to vortex shedding and flow excited resonances at the standpipe inlet and interfering with the SRV operation. The responses were as follows:

(a) The SRVs for almost all plants (except a few early plants) are located on the top of the horizontal run of the main steam line piping with the standpipe oriented vertically. In this orientation, the inlet to the side branch located at the bottom of the standpipe prevents the accumulation of condensate in the standpipe. This orientation will also ensure that loose parts cannot collect in the standpipe. The shear layer instability and vortex shedding associated with flow-excited acoustic resonances of the standpipe are local to the entrance region of the standpipe. The vortex acoustic coupling produces an in-out flow with rapid cycling (typically over 100 Hertz for a relief valve), and so the likelihood of drawing a loose part in without it subsequently being driven out into the main steam line flow in the next half cycle is very small. In order for the loose part to be drawn into the standpipe region, the loose part would have to be traveling along the top of the steam line past the standpipe opening, pass by the opening at the right time in the vortex cycle to be subjected to an upward drag force to lift the loose part up and out of the main flow stream. It is unlikely that anything other than the very small pieces could be lifted up and out of the main stream and carried into the standpipe by the vortex. Once inside the standpipe, there is no force that will hold the loose piece in the standpipe where the fluid is stagnant. The cyclical nature of the vortex will tend to return the loose part to the main flow stream. Therefore, the NRC staff finds that it is unlikely that the loose parts will be drawn into the standpipe by vortex shedding associated with flow-excited acoustic resonances in such a way that it can interfere with the SRV operation.

(b) A few of the early plants may have SRVs mounted on a vertical run of the main steam line with the standpipe oriented horizontally. In this configuration, it is possible that a small loose part may come to rest in the horizontal section of the standpipe. Again, in order for the shear layer instability and vortex shedding associated with flow-excited acoustic resonances to affect the trajectory of the loose part, the loose part would have to be traveling along the side of the steam line right next to the standpipe opening, at just the right time in the vortex cycle, and the part would have to be very small in order for the vortex to be able to deflect the part out of the main flow stream. It is unlikely that these conditions will be satisfied and, therefore, it is unlikely that the vortex shedding associated with flow-excited acoustic resonances will draw a loose part into the standpipe where it can interfere with the SRV operation. The BWRVIP stated that the momentary occurrence of a passing loose part coincident with an AOO is incredible. The NRC staff finds the BWRVIP rationale on the unlikely nature of loose parts interfering with the SRV operation to be acceptable.

Based on this evaluation, the NRC staff has determined that, regarding structural issues, the BWRVIP has adequately considered safety and operational aspects in addressing the impact of loose parts.

3.2 Loose Parts and Components Issues

After reviewing the information provided in the revised Section 4.0 of TR BWRVIP-06-A, the NRC staff requested the BWRVIP to submit additional information. The response to the RAI was provided by letter dated November 30, 2007 (ADAMS Accession No. ML073410041).

3.2.1 Standby Liquid Control System Standpipe

In Sub-section 4.1 of the revised Section 4.0 of TR BWRVIP-06-A, the potential impact of loose parts on the SLC system standpipe and the safety consequences were not discussed. Therefore, the NRC staff issued a RAI on the subject. In response, the BWRVIP stated that the DP and the SLC line consist of two sections: the first one is the SLC line section serving as a pressure tap for one end of the core DP instrumentation and the second pressure tap is an instrumentation line running to the top of the core plate. The DP and the SLC line are supported at the vessel nozzle, the shroud support skirt, and along the shroud as it is routed inside the core. The resulting configuration is a pipe within a pipe that is well supported and is unlikely to be damaged from impact by a loose part entering the lower plenum. It was further stated that even if the pipe attached to the core shroud was damaged by a loose part, it would not prevent the injection of sodium pentaborate into the vessel. This pipe is for core DP measurement and not standby SLC. Additionally, failure of the pipe in the lower plenum would result in a change in measured core DP and be detected in the control room. Therefore, it was concluded that there is no safety concern associated with the potential impact on the SLC system standpipepipe due to the presence of potential loose parts, and that the item need not be added in the list of concerns. The NRC staff finds the response acceptable.

3.2.2 Main Steam Isolation Valves

In Section 4.1.1 of the revised Section 4.0 of TR BWRVIP-06-A, potential interference by loose parts with MSIVs was discussed. The BWRVIP indicated that the fixed liner on the vertical centerline could be bent or broken when hit by a large loose part that might prevent the valve from closing. The NRC staff believed that there is a potential for both MSIVs in a single steam line to fail to close, and therefore, requested that the BWRVIP address the possibility for both valves failing to close due to large loose parts generated as a result of the steam dryer failure. In response, it was stated that there is no known case where a redundant MSIV has failed to provide the design backup expected. BWRs are designed with redundancy as a standard for valves in a line that require the redundancy. It is highly unlikely that two large steam dryer loose parts would migrate down to a single main steam line and bend or break both of the legs on the vertical centerline of both valves. Two valves were considered adequate for the original design, with approval being given based on the necessity of a two valve redundant design. One or more large potential loose parts from a failed steam dryer should be detected by change in steam moisture quality. In addition, if a large loose part were to become lodged in a main steam line this would likely be detected as a mismatch in flow by control room indicators. As a result, the probability of failure of both MSIVs to close in the same steam line due to damage caused by loose parts is insignificant. The NRC staff finds the response acceptable.

3.2.3 Safety/Relief Valve Operation

In sub-section 4.1.2 of the revised Section 4.0, the BWRVIP indicated that potential loose parts are not expected to interfere with the SRV operation because of the following reasons:

(1) SRVs are closed during normal plant operation, and as such there is no flow to draw the loose parts into the standpipepipe and the valve, and (2) the failure of system components coinciding with a loss-of-coolant accident is unlikely. Recent industrial experience has suggested that the shear layer, which separates the mean flow in the main pipe from the stagnant fluid in the branch, can be unstable due to the acoustic resonance in the SRV standpipe. The NRC staff requested the BWRVIP to address the possibility that the loose parts may be drawn into the standpipe and the valve due to vortex shedding and flow-excited acoustic resonances at the inlet. The BWRVIP responded by stating that the SRVs for almost all plants are located on the top of the horizontal run of the main steam line piping with the standpipe oriented vertically. This orientation prevents the accumulation of condensate in the standpipe. This orientation will also ensure that loose parts cannot collect in the standpipe. The shear layer instability and vortex shedding associated with flow-excited acoustic resonances of the standpipe are local to the entrance region of the standpipe. The vortex acoustic coupling produces an in-out flow with rapid cycling (typically, over 100 Hertz for a relief valve), and so the likelihood of drawing a loose part in without it subsequently being driven out into the main steam line flow in the next half-cycle is very small. In order for the loose part to be drawn into the standpipe region, the loose part would have to be traveling along the top of the steam line past the standpipe opening, pass by the opening at the right time in the vortex cycle to be subjected to an upward drag force, and the vortex would have to be strong enough to lift the loose part up and out of the main flow stream. Given the flow velocities in the main steam line, it is unlikely that anything other than the very smallest pieces could be lifted up and out of the main stream and carried into the standpipe by the vortex. Once inside the standpipe, there is no force that will hold the loose piece in the standpipe where the fluid is stagnant; the cyclical nature of the vortex itself will tend to return the loose part to the main flow stream. Therefore, it is unlikely that the loose parts may be drawn into the standpipe by vortex shedding associated with flow-excited acoustic resonances in such a way that it can interfere with the SRV operation.

The response further stated that a few of the early plants may have SRVs mounted on a vertical run of the main steam line with the standpipe oriented horizontally. In this configuration, it is possible that a small loose part may come to rest in the horizontal section of the standpipe. However, in order for the shear layer instability and vortex shedding associated with flow-excited acoustic resonances to affect the trajectory of the loose part, the loose part would have to be traveling along the side of the steam line right next to the standpipe opening, at just the right time in the vortex cycle, and the part would have to be very small in order for the vortex to be able to deflect the part out of the main flow stream. It is unlikely that these conditions will be satisfied and, therefore, it is unlikely that the vortex shedding associated with flow-excited acoustic resonances will draw a loose part into the standpipe where it can interfere with the SRV operation. The NRC staff finds the response acceptable.

3.2.4 High Pressure Coolant Injection and Reactor Core Isolation Cooling

Sub-section 4.1.7 of the revised Section 4.0 discusses the potential for interference with the HPCI and RCIC operation, indicating that during normal operation, both systems are idle and stagnant, and no flow will draw loose parts into the system piping. Similar to the previous RAI, the NRC staff requested that the BWRVIP address whether loose parts could be drawn into the system piping when the shear layer at the opening to the HPCI and RCIC systems become unstable due to a higher flow of the main steam line because of acoustic resonance. The BWRVIP responded by stating that in order for the shear layer instability and vortex shedding associated with flow-excited acoustic resonances to affect the trajectory of the loose part, the loose part would have to travel along the side of the steam line right next to the HPCI or RCIC opening, at just the right time in the vortex cycle, and the part would have to be very small in order for the vortex to be able to deflect the part out of the main flow stream. It is unlikely that these conditions will be satisfied and, therefore, it is unlikely that the vortex shedding associated with flow-excited acoustic resonances will draw a loose part into the HPCI or RCIC where it can interfere with the HPCI or RCIC operation. The NRC staff finds the response acceptable.

3.2.5 Upper and Lower Tie Geometry

Sub-section 4.1 of the revised Section 4.0 stated that, "The geometry of certain parts and their components may be such that they would not be able to pass through the fuel bundle upper tie plate openings." Furthermore, while discussing potential for interference with control rod operation, sub-section 4.1.4 of the TR stated that, "The debris filter on the lower tie plates could stop even the smallest parts." The NRC staff requested that the information regarding the fuel bundle upper tie plate openings and the mesh size for the debris filter on the lower tie plate be provided. In response, the BWRVIP provided upper tie plate pass through sizes for potential loose parts with rectangular, cylindrical, and square shapes for various GE Nuclear Fuel BWR fuel designs and other fuel designs (Westinghouse and AREVA) currently in use in BWRs. In addition, debris filter mesh opening sizes for GE fuel and other fuel designs (Westinghouse and AREVA) currently in use in BWRs were provided.

Based on the information provided, the NRC staff believes that installing debris filters will limit the size of the loose parts to pass through. However, the NRC staff agrees with BWRVIP that there is still a safety concern associated with the potential of a control rod to scram due to the presence of potential smaller loose parts. In the worst-case scenario, the accident analysis covers the condition of loose parts causing a single failure to scram a control rod. Therefore, this condition is unlikely to negatively affect safe shutdown and offsite dose.

3:2.6 Loose Parts History

The NRC staff noted that some licensees keep a list of loose parts from the beginning of operation of the plant. The NRC staff, therefore, recommended that the BWRVIP add a section in the TR to indicate that keeping an inventory and history of the loose parts (indicating the number, size, date the part was lost, and the date it was recovered) is a good practice to keep a track of the loose parts, per RAI 06-A-13. The BWRVIP agreed to modify revised Section 4.0 of BWRVIP-06-A to include statements to reflect that it is a good practice to keep an inventory and history of the loose parts including the number, the size, the date the part was lost, and the date it was recovered.

3.2.7 Regulatory Guide 1.133

The NRC staff requested the BWRVIP to reference RG 1.133, Revision 1, "Loose Parts Detection Program for the Primary Systems of Light Water Cooled Reactors," 1981, in the revised Section 4.0 of the BWRVIP-06-A TR. RG 1.133 describes a method acceptable to the NRC staff with respect to detecting a potentially safety-related loose part in light-water-cooled reactors during normal operation by installing a Loose Parts Monitoring System (LPMS). In response, the BWRVIP indicated that referencing RG 1.133 in the revised Section 4.0 of the BWRVIP-06-A TR would lead to confusion because a review of the operating history of the LPMS does not indicate significant differences in the impact or consequence of loose parts in the reactor coolant pressure boundary between plants with a LPMS, and those without. In an NRC letter dated January 25, 2001 (ADAMS Accession No. ML051110043), from S.A. Richards to J.M. Kenny (BWROG), "BWR Owners Group - Topical Report NEDC-32975(P), Regulatory Relaxation For BWR Loose Parts Monitoring Systems," the NRC approved the regulatory relaxation of LPMS that were requested by the BWROG, TR NEDC-32975P, "Regulatory Relaxation for BWR Loose Parts Monitoring Systems," dated July 31, 2000 (ADAMS Accession No. ML003754771), and subsequent removal of the LPMS from other operating plants. The NRC staff approval and the associated design evaluation that defines the basis for NRC acceptance of regulatory relaxation of LPMS is available in ADAMS as Accession No. ML051110043. The NRC staff concluded that the safety benefits of the LPMS do not appear to be commensurate with the cost of maintenance and the associated radiation exposure for the plant personnel. The NRC staff agrees with BWRVIP and accepts that referencing RG 1.133, Revision 1, in TR BWRVIP-06-A is not necessary.

3.3 Radiation Release Limits

Sub-section 4.1.1.1 of the revised Section 4.0 specified 10 CFR Part 100 limits for off-gas radiation release. The NRC staff believed that the main concern that should have been addressed is as low as reasonably achievable (ALARA), and hence 10 CFR Part 20, "Standards for Protection Against Radiation," is applicable. The NRC staff requested that the BWRVIP address the application of 10 CFR Part 20 with regards to ALARA considerations. In response, the BWRVIP stated that the subject statement concerns the ability of the off-gas system to monitor and maintain off-site releases below plant-specific licensing limits to comply with 10 CFR Part 100 requirements. This is because one of the evaluation bases of the BWRVIP-06-A TR was "The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10CFR100 guidelines." As a result, 10 CFR Part 100 was addressed in the TR.

The response further stated that all facilities are required to follow 10 CFR Part 20, "Standards for Protection Against Radiation," and are required to have an approved ALARA program for work at the facility, and it is expected that all work at the facility follows a procedure requiring that a necessary radiation survey to be completed prior to the start of any work in a radiation environment. With a completed radiation survey, the worker is expected to keep the work radiation exposure absorbed ALARA. In an ALARA program, the following four major ways are relied upon to reduce radiation exposure from any radioactive loose part to workers or to the population:

- Shielding: Use proper barriers to block or reduce ionizing radiation,
- Time: Spend less time in radiation fields,
- Distance: Increase distance between radioactive sources and workers or population, and
- Amount: Reduce the quantity of radioactive material for a practice.

Because plants are required to have an approved ALARA program, and are required to follow the program to minimize radiation exposure, the NRC staff finds the above response acceptable.

3.4 Loose Parts and Corrosion Issues

Section 4.1.6 of the revised Section 4.0 of TR BWRVIP-06-A addresses potential corrosion that could be caused in the RVIs by the presence of loose parts. The NRC staff agrees with the BWRVIP's contention that corrosion of RVI components due to loose parts is not a concern because the materials used for the RVI components are the same as the materials in loose parts. Consequently, general corrosion should not take place in RVIs. However, localized crevice corrosion can occur as a result of a potential crevice condition caused by a loose part lodged on top of any RVI. By letter dated December 21, 2006 (ADAMS Accession No. ML070030188), the NRC staff requested that the BWRVIP address possible crevice corrosion due to the presence of a crevice created between a lodged loose part and the RVI. Crevice corrosion can be more pronounced in RVIs where effective protection due to hydrogen water chemistry and/or noble metal chemical addition is not available and when the reactor water flow is stagnant. By letter dated November 30, 2007 (ADAMS Accession No. ML073410041), the BWRVIP stated that the tight deep crevices with susceptible microstructures are more prone to crevice corrosion when they are subject to applied stresses during normal operating conditions. The BWRVIP reiterated that plant experience indicates that there are many examples where one of these factors is present and yet no crevice corrosion has been observed. For example, a bolted structure does have a crevice at the thread root location. However, in the absence of any susceptible microstructure (sensitized) such as the one that is typically present in the heat affected zone of a weld joint, thus far, no crevice corrosion was observed. A postulated loose part may not be susceptible to crevice corrosion mainly due to the absence of stress or susceptible microstructure or cold work. Additionally, an existing loose part may not have a tight crevice because it is not mechanically attached to a RVI. Based on the above, the BWRVIP concluded that there is no concern regarding the onset of crevice corrosion in RVI components as a result of a loose part.

The NRC staff reviewed the BWRVIP response and concludes that the BWRVIP response is acceptable because: (1) plant experience does substantiate the claim that, thus far, no failures due to crevice corrosion has occurred in non-welded areas (i.e., bolts); (2) absence of any tight crevice between the loose part and the RVI minimizes the chances of crevice corrosion; (3) no susceptible microstructure exists in a loose part as it does not contain any weld and consequently, no sensitized heat affected zone exists and, (4) lack of crevice corrosion in non-welded RVIs (i.e., bolts) suggest that effect of cold work may not play a major role in crevice corrosion.

Based on the above evaluation, the NRC staff concludes that it is less likely that crevice corrosion will occur in RVI components as a result of loose parts. Therefore, the NRC staff considers that its concern is resolved when the BWRVIP includes its RAI response related to the corrosion issue in the "A" version of the revised Section 4.0 of TR BWRVIP-06-A.

4.0 LIMITATIONS AND CONDITIONS

When referencing the revised Section 4.0 of TR BWRVIP-06-A in licensing applications, the following limitations and conditions shall apply:

1. The conclusions of the safety assessment documented in the revised Section 4.0 of BWRVIP-06-A report shall apply only to BWR/2-6 plants.

2. Because the evaluation provided in TR BWRVIP-06-A is general in nature and does not assess the effects on the basis of individual geometries or material properties, as these are resolved on a case-by-case basis, and because several plant-specific features can determine the acceptable loose part size limits for safety and operational concerns, TR BWRVIP-06-A and the SE for revised Section 4.0 provide only general criteria for acceptability of loose parts in operating BWR/2-6 plants. Plant-specific safety assessment is, therefore, required to be performed by a licensee in the event loose parts are detected in its plant.

3. The NRC staff considers that its concern related to corrosion is resolved when the BWRVIP includes its RAI response related to the corrosion issue discussed in Section 3.4 of this SE in the "A" version of the revised Section 4.0 of TR BWRVIP-06-A.

4. The BWRVIP agreed to modify revised Section 4.0 of BWRVIP-06-A to include statements to reflect that it is a good practice to keep an inventory and history of the loose parts including the number, the size, the date the part was lost, and the date it was recovered in the "A" version of the TR.

5.0 CONCLUSION

The BWRVIP evaluation as described in revised Section 4.0 of TR BWRVIP-06-A shows that safe reactor operation and safe shutdown capability are not compromised for most categories of postulated loose part sizes. Based on the operating experience of BWRs, larger and heavier parts like pieces of a jet pump beam or plate from a steam dryer have not negatively affected safe shutdown or offsite dose. While it is possible for loose parts, with particular size and shape, to compromise fuel performance, it is extremely unlikely that such loose parts would result from the failure of RVIs. If it were to occur, damage would likely be limited to a single fuel bundle, which would be detected by routine monitoring of the off-gas monitors or LPRMs. As a result, there is no significant safety concern from potential loose parts on fuel. In addition, there is no safety concern for interference with MSIVs, control rod operation, damage to reactor internals, corrosion or chemical reaction with other reactor materials, interference with HPCI or RCIC operation, RWCU or RHR isolation valves, nuclear instrumentation, and RHR pumps and heat exchangers. There could be some possible operating concerns from the potential loose part(s) with regard to fuel fretting, bottom head drain plugging, and recirculation system performance, but none of these are expected to negatively affect safe shutdown or increase in off-site dose.

The NRC staff concludes that the generic safety assessment provided in revised Section 4.0 of BWRVIP-06-A is applicable only for BWR/2-6 reactor internals. The evaluation provided in the TR is general in nature and does not assess the effects on the basis of individual part geometries or material properties, as these are resolved on a case-by-case basis. In addition, several plant-specific features can determine the acceptable loose part size limits for safety and operational concerns. As a result, the NRC staff also concludes that this SE provides only general criteria for acceptability of loose parts in operating BWR/2-6 plants, and that a plant-specific safety assessment is required to be performed by the licensee in the event of loose parts being detected.

The NRC staff, therefore, finds the revised Section 4.0 of the BWRVIP-06-A TR acceptable for referencing, subject to the limitations and conditions identified in this SE.

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RECORD OF REVISIONS

Revision Number	Revisions
BWRVIP-06	Original Report (TR-105707)
BWRVIP-06-A	The report as originally published (TR-105707) was revised to incorporate changes proposed by the BWRVIP in responses to NRC Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE), and other necessary revisions identified since the last issuance of the report. *No changes to the report were required.* In accordance with a NRC request, the NRC SE is included here as an appendix and the BWRVIP report number includes an "A" indicating the version of the report accepted by the NRC staff. Non-essential format changes were made to comply with the current EPRI publication guidelines. Details of the revision can be found in Appendix E.
BWRVIP-06, Revision 1-A	A previous version of this report was published as BWRVIP-06-A (1006598) and revised by replacing Section 4.0 "Consideration of Loose Parts" in its entirety. This report (BWRVIP-06, Revision 1-A, EPRI 1019058) incorporates changes proposed by the BWRVIP in response to U. S. Nuclear regulatory Commission (NRC) Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE) and other necessary revisions identified since the previous publication of the report. All changes except typographical errors are marked with margin bars. GEH proprietary information is designated by underlining. Non-essential format changes were made to comply with the current EPRI publication guidelines. In accordance with NRC guidance, the NRC SE, as well as other NRC correspondence has been included. NRC Safety Evaluation added to front matter. New section 1.4 "Implementation Requirements" added. Section 4.0 "Consideration of Loose Parts" replaced in its entirety. Old Appendix B "Record of Revisions" deleted and information included in new Appendix E "Record of Revisions". New Appendices B-D added providing additional NRC correspondence. Details of the revision can be found in Appendix E.

EXECUTIVE SUMMARY

This report provides a safety assessment addressing potential failures due to cracking of the internal components in BWR/2 through BWR/6 product lines. All components were reviewed to determine short-term and long-term actions which are appropriate to assure continuing safe operation given the possibility of cracking in those components.

The assessment considers the component safety function during normal operation and in response to design basis accidents, transients and seismic events. The safety functions considered are those associated with (1) maintaining coolable geometry, (2) maintaining control rod insertion times, (3) maintaining reactivity control, (4) assuring core cooling and (5) assuring instrumentation availability. Within each component, individual failure locations (welds or bolted connections) were identified and the consequence of failure at those locations, including the impact of loose parts, was considered.

The results of the evaluation support the conclusion that no short-term action is needed; all BWR product lines possess a sufficient level of safety, based on one or more of the following considerations:

- Detectability of component failure by on-line instrumentation or current inspections.
- Structural redundancy within a component.
- Low probability of a challenging event.

In the long term, some potential cracking locations in components whose function is needed to respond to design basis LOCA, ATWS or seismic events are identified as requiring actions. Such actions could include inspection, assessment, repair or mitigation. Actions are currently underway at many utilities to address degradation of some of these components. Where actions are identified as required, the BWR Vessel & Internals Project (BWRVIP) will be developing the necessary elements of inspection, assessment, mitigation or repair from which individual utilities can develop plant specific-reactor internals management programs.

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1
INTRODUCTION

1.1 Purpose

This document provides safety assessments of BWR reactor internals and attachments in order to determine short-term and long-term actions required to assure safe operation given the possibility of component cracking. Where actions are identified, the BWR Vessel & Internals Project (BWRVIP) will be establishing plans to develop the necessary elements of inspection, assessment, mitigation or repair. Utilities will then be able to tailor a reactor internals management program from the BWRVIP elements to assure continuing safe operation.

In addition, this document is intended to provide utilities and regulatory agencies with information needed to evaluate the impact of future postulated or observed cracking in BWR reactor pressure vessel (RPV) internal components.

1.2 Scope

This document provides safety assessments assuming loss of integrity in welded or bolted locations in reactor internal components. The assessments include descriptions of component function and the variations in design among the BWR/2-6 plants. Then, for each component, there is a general discussion of consequences of potential cracking. In addition, the specific consequences of potential cracking at each welded or bolted location are discussed.

The discussions of component variation are based on the original component designs. There may be plant-specific variations which have resulted from modifications during construction or operation. In general, these modifications represented improvements or were designed to restore integrity to a degraded component. However, it is the responsibility of the utility to confirm that the evaluations in this report are bounding for their modified components.

Evaluations are provided for all reactor internal components, with the boundary of evaluation being the attachment or penetration weld connecting the internal component to the vessel. The vessel and other pressure boundary components are not addressed here, as they are by ASME Code inspection, evaluation and repair requirements. Components which are consumable, such as fuel bundles, control rods and incur instruments, are not considered in this document. Table 1-1 identifies the components evaluated, their safety category and the section where they are addressed. Figures 1-1 and 1-2 show an overview cutaway of a typical BWR and the locations of the reactor internals.

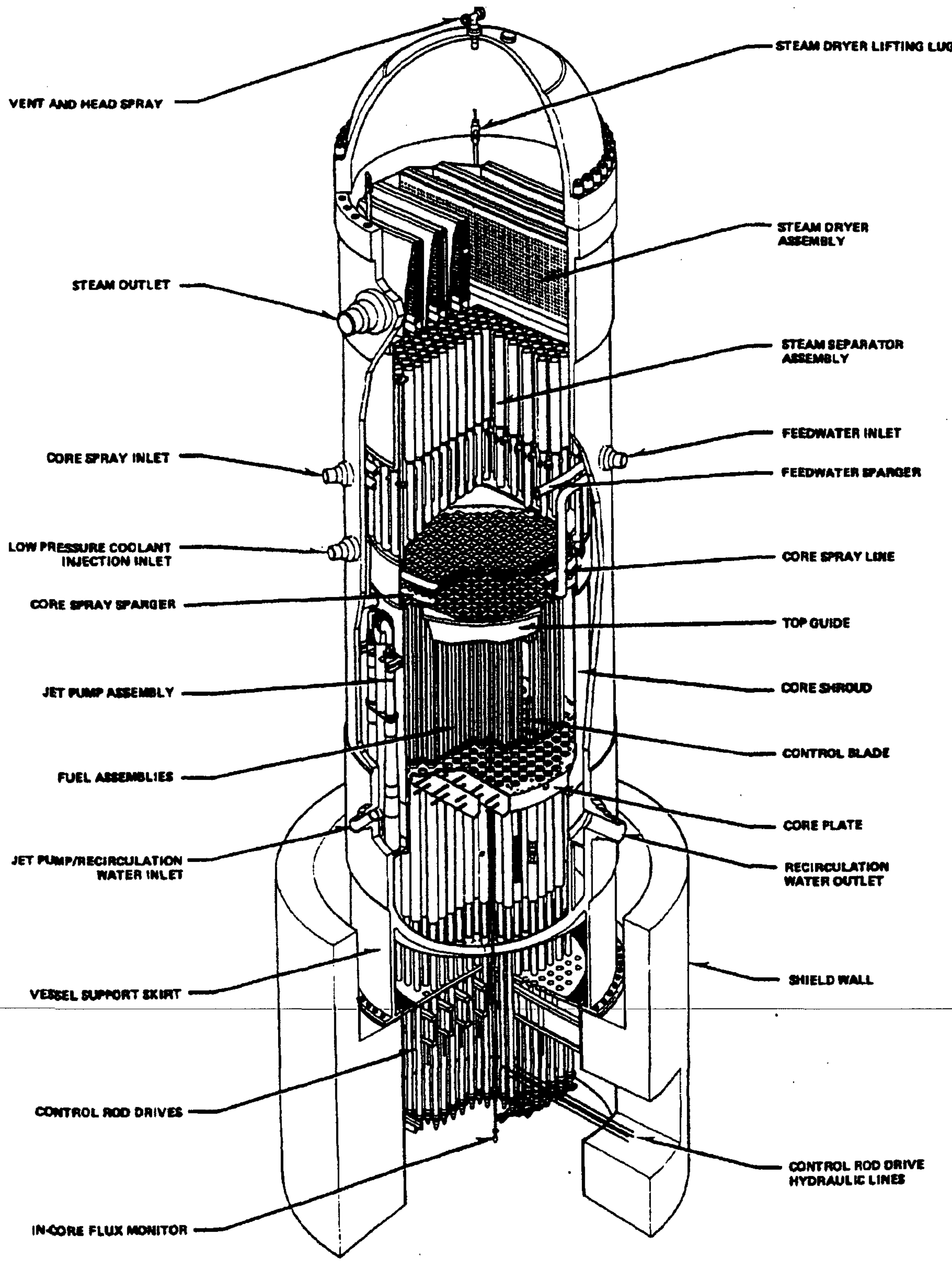

Figure 1-1
Reactor assembly typical jet pump BWR/6 shown

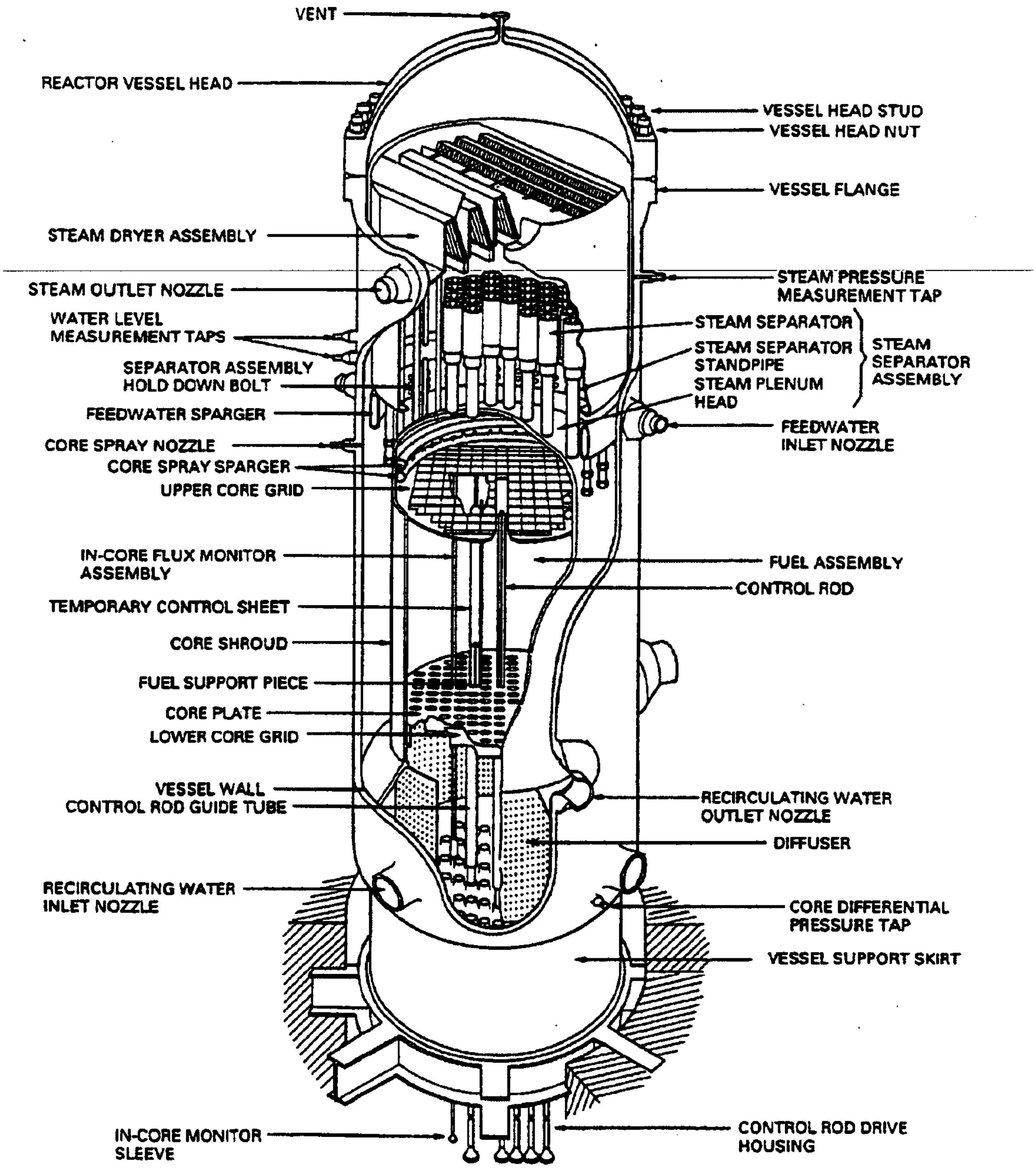

Figure 1-2
Reactor assembly typical non-jet pump BWR/2

Section 2 of this document addresses components which are categorized as safety-related. Each component in Section 2 is evaluated to the extent described above. Sketches provided in Section 2 provide typical component information. Plant-specific variations may exist. GE proprietary information in Section 2 is denoted by margin bars.

Section 3 discusses non-safety-related components. The evaluation for each of these components is less extensive, as it is intended only to explain the basis for each component's non-safety-related categorization, as well as consequences of component failure.

Section 4 discusses the potential safety consequences of loose parts. It is not feasible to evaluate every possible loose part that could conceivably originate at each internal component. Instead, the range of loose parts scenarios is bounded by evaluating the safety consequences of hypothetical loose parts of various sizes, originating from various regions of the reactor. These evaluations apply to loose parts generated by both safety-related and non-safety-related components.

1.3 Evaluation Basis

For purposes of evaluation, specific internal component locations, such as welds or bolts, are assumed to be fully failed and the resulting safety consequences are evaluated. Bolts and welds are postulated as failure locations because historically these are the locations most susceptible to cracking. Multiple cracks are considered only to the extent that they can be reasonably postulated by engineering judgment. The bounding characteristics of a location failure are assumed to assure that the assessment is conservative. For example, complete separation is assumed if that condition is more bounding than leakage from a partial failure.

As an evaluation starting point, the components which are categorized as "safety-related" are those that can be relied upon to remain functional during and following design basis events to ensure:

1. The integrity of the reactor coolant pressure boundary.
2. The capability to shut down the reactor and maintain it in a safe shutdown condition.
3. The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10CFR100 guidelines.

Evaluation of the internal components, because they are inside the reactor, can more easily be focused on the functions. All reactor internal components are passive and their general safety function is to remain structurally sound. The specific safety function of each safety-related reactor internal component is discussed in Section 2; in general, however, they are a mix of the following functions:

- Maintaining a coolable geometry.
- Maintaining control rod insertion times.
- Maintaining reactivity control.
- Assuring core cooling effectiveness.
- Assuring instrumentation availability.

Design Basis

Failures discussed in this document are beyond the current design or licensing basis of operating BWRs. All reasonably postulated events are considered. Their consideration is solely for the purpose of establishing the relative importance of postulated failures. No new design bases are implied by the failures considered in this document.

Evaluation Assumptions

For this evaluation, the worst case assumption of complete failure at a component location is being made. Therefore, the acceptance criterion is to achieve safe shutdown, not to maintain original design margins. All realistically available systems are considered in this evaluation regardless of safety classification. The evaluation of a failure location focuses on its impact on accident consequences and plant responses. Within these criteria, a failure location in some safety-related components could be accommodated without significant concern for achieving safe shutdown, based on some of the component characteristics discussed below:

1. **Detectability** - If a location failure which could interfere with safe shutdown during an accident scenario can be detected during plant operation, a safe shutdown can be achieved before the component is challenged by the accident scenario. (*Note*: in some cases, this may require implementation of new procedures to more closely monitor reactor operating conditions and compare them to an established "normal baseline").

2. **Redundancy** - If failure of a component location results in the loads being redistributed to other components or locations on the same component which have adequate margin to accommodate the additional loads, there will be no adverse impact on the ability of that component to function in achieving safe shutdown.

   Degradation mechanisms have a significant randomness, such that the degree of cracking in different locations would be varied, and complete failures would not occur simultaneously. Historical experience supports the conclusion that structurally redundant components, redundant locations within a component, or systems with redundant functions are expected to have enough degradation randomness that simultaneous failures would not occur.

3. **Inspection** - If inspection is being performed with sufficient frequency and detail, the possibility of significant undetected cracking can be excluded when considering the short-term significance of potential cracking.

Table 1-1
RPV internal components

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1.4 Implementation Requirements

In accordance with the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues, all sections of this report are "for information only" with the exception of the statement in Section 4.0 that it is a good practice to maintain an inventory and history of lost parts which is a "good practice".

2 SAFETY-RELATED COMPONENT SAFETY ASSESSMENTS

2.1 Control Rod Guide Tube, CRD Housing and Stub Tube

2.1.1 Hardware Evaluation

Component Description and Function

The control rod guide tubes, control rod drive (CRD) housings and stub tubes provide an assembly of components (Figure 2-1) at symmetrical locations below the core which support the weight of the fuel (except some peripheral bundles supported by the core plate) and allow the movement of control rods into the reactor core to achieve reactivity control. Control rods provide the primary means of achieving shutdown conditions.

The control rod guide tube (Figure 2-2, 2-3) extends from above the core plate to the CRD housing which extends from the vessel bottom head. The guide tube is mechanically connected to the housing by a bayonet mechanism which engages the thermal sleeve inside the housing. The guide tube is held in a locked position by anti-rotation pins on the core plate, which engage slotted alignment lugs welded to the top of the guide tube. The guide tube houses the control rod when it is withdrawn from the core, and has holes just below the core plate to provide a flow path for water in the vessel bottom head to enter the core. The fuel support casting, discussed in Section 2.9, is inserted into the top of the guide tube and has holes aligned with the guide tube holes to direct core flow to the individual fuel bundles.

The CRD housing contains the CRD mechanism for controlling the position of the control rods. On pre-BWR/6 plants stub tube penetration Figure 2-4 at the reactor pressure vessel boundary supports the loads from the CRD housing caused by vessel internal pressure and the weight of the fuel, and the stub tube is welded to the vessel bottom head. BWR/6 plants (Figure 2-4) do not normally contain a stub tube; loads from the CRD housing are transferred directly to the bottom head, however, some BWR/6s do have stub tubes.

The CRD housings extend several feet outside the vessel bottom head, becoming part of the vessel pressure boundary. The CRD mechanisms are mounted to the end of the housing by a bolted flange connection.

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Figure 2-1
Control rod guide tube, housing and stub tube assembly

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Figure 2-2
Assembled drive line

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Figure 2-3
Control rod guide tube BWR/2, 3, 4, 5, 6

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Figure 2-4
Control rod drive housing

Failure Locations and Product Line Variations

Figures 2-1 to 2-4 show the component details for the different product lines. Table 2-1 summarizes the potential failure locations and product line variations.

Table 2-1
Potential guide tube & housing assembly failure locations

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2.1.2 Safety Assessment*

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Location 1 – Guide Tube to Alignment Lug and Anti-Rotation Pin

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Locations 2, 3, and 3a – Guide Tube and Control Rod Housing Circumferential Welds

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Locations 4 – CRD Housing to Stub Tube Weld and 4a – CRD Housing to Bottom Head Weld

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Location 5 – Stub Tube to Bottom Head Weld

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2.1.3 Conclusions and Actions

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2.2 Core Plate dP/Standby Liquid Control (SLC) Line

2.2.1 Hardware Evaluation

Component Description and Function

The Standby Liquid Control (SLC) System is designed to shut down the reactor from full power by injecting a neutron absorber (sodium pentaborate) into the reactor core when the normal method of controlling core reactivity with control rods cannot be accomplished. In most plants, a line from the SLC nozzle/penetration in the vessel bottom head supplies liquid sodium pentaborate to a standpipe or sparger inside the RPV, which, in turn, distributes the liquid through holes to the coolant entering the core. In some plants, the SLC is supplied to the core through the core spray line (Section 2.4).

A line within the SLC injection line, or a separate line in the bottom head (BWR/6), is used in the detection of differential pressure across the bottom core plate. The core plate dP line instrumentation provides information on core flow performance for diagnostic purposes, on CRD system water differential pressure indication and on core spray piping break detection.

Failure Locations and Product Line Variations

Figures 2-5 through 2-8 show the RPV internal piping arrangements for the core plate dP/SLC lines for various BWR product lines, along with the potential failure locations of the welds. Table 2-2 identifies the potential failure locations and product line applicability.

Table 2-2
Potential SLC/core plate dP failure locations

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Figure 2-5
Differential pressure and liquid control line BWR/2

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Figure 2-6
Differential pressure and liquid control line BWR/3, 4

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Figure 2-7
Differential pressure and liquid control line BWR/5

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Figure 2-8
Differential pressure and liquid control line BWR/6

2.2.2 Safety Assessment*

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Locations 4 and 6 – SLC Standpipe or Sparger Welds

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Locations 1 and 5 – Above Core Plate Sensing Line Welds

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2.2.3 Conclusions and Actions

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Long Term

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2.3 Core Plate

2.3.1 Hardware Evaluation

Component Description and Function

The core plate provides lateral support for the fuel bundles, control rod guide tubes and in-core instrumentation during seismic events and provides vertical support for the peripheral fuel assemblies. The core plate assembly (Figure 2-9) consists of a perforated stainless steel plate reinforced by stiffener beams and supported on the perimeter by a circular rim. The core plate rim is bolted to a ledge on the core shroud by stainless steel studs which prevent vertical movement.

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Figure 2-9
BWR core plate

Stiffener beams are welded to the core plate to carry the pressure loads following design basis LOCA events. Because peak pressure loads place the lower edges of the beams in compression, cross ties between the beams are required for stabilization. The stabilizer beams or rods prevent flange buckling by providing lateral support. These beams or rods also provide support for in-core housing monitors.

Failure Locations and Product Line Variations

The core plate designs are similar in all product lines. Table 2-3 summarizes the potential failure locations. Core plates are positioned on the shroud ledge by four vertical or horizontal aligner pins (not applicable to BWR/6 plants), which vary in configuration from plant to plant. The pin assembly engages bosses or sockets which are welded to both the core plate and the shroud.

Table 2-3
Potential core plate failure locations

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In the case of BWR/2 through BWR/5 plants, seismic and other dynamic loads are shared between the friction load of the shroud to rim bolt connection, and the shear resistance of the aligner pins. BWR/6 plants were designed to be restrained by wedges and studs between the core plate rim and the shroud.

Failure locations are identified on Figures 2-9 through 2-21.

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Figure 2-10
BWR/2 core plate

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Figure 2-11
Peripheral fuel support to core place weld (2)

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Figure 2-12
Core plate bolt (10) typical for BWR 2-5

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Figure 2-13
Block aligner on pin (8) typical for BWR 2-5 with vertical aligners

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Figure 2-14
BWR-3 core plate

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Figure 2-15
BWR-4 core plate type 1

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Figure 2-16
BWR-4 core plate type 2

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Figure 2-17
BWR-4 core plate type 3

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Figure 2-18
Core plate plugs and BWR-4 horizontal aligner arrangement

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Figure 2-19
Core plate for a BWR-5

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Figure 2-20
BWR/6 core plate

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Figure 2-21
BWR-6 core plate wedge

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Figure 2-22
BWR-6 core plate bolt (10)

2.3.2 Safety Assessment*

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Locations 1, 4 and 5 – Stiffener and Stabilizer Beam Welds

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Location 2 – Peripheral Fuel Support to Core Plate

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Location 3 – Stiffener Beam to Rim Welds

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Location 6 – In-core Guide Tube Support to Stabilizer Beam

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Location 7 – Core Plate to Rim Weld

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Location 8 – Aligner Pin, Socket to Rim and Socket to Shroud Welds

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Location 9 – Core Plate Wedge Retainer

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Location 10 – Rim Holddown Bolts

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Location 11 – Rim Fabrication Welds

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Location 12 – Core Plate Fabrication Welds

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Location 13 – Core Plate Plug

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2.3.3 Conclusions and Actions

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2.4 Core Spray Piping

2.4.1 Hardware Evaluation

Component Description and Function

The core spray piping internal to the RPV provides the flow path of core cooling water from the vessel nozzle, through the shroud to the core spray spargers (Section 2.5) above the core region. Figure 2-23 shows the overall core spray arrangement. At some BWRs either the high pressure or low pressure core spray piping also is used to provide a pathway for the SLC System to inject sodium pentaborate directly into the core region. Core spray and other makeup system combinations, by plant type, are shown in Table 2-5.

The core spray piping is supported by brackets which are bolted to bosses which are welded to the vessel wall. The bracket, bolt and boss assembly reacts loading from weight, seismic events and pressure transients.

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Figure 2-23
Typical core spray piping configuration

Table 2-4
Potential core spray piping weld failures

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Table 2-5
BWR coolant make-up systems

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XX	2 or 3 Loop Systems	*	Available at some plants	+	Non-safety System
X	Single Loop Systems	(1)	Some plants inject SLC above the core plate		
		(2)	Some plants inject a portion of HPCI flow inside the shroud		

Failure Locations and Product Line Variations

Figures 2-23 through 2-27 show potential failure locations in typical core spray piping. Table 2-4 indicates the potential failure locations and product line applicability.

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Figure 2-24
BWR/2 core spray piping configuration

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Figure 2-25
Core spray piping and spargers inside RPV (typical)

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Figure 2-26
Core spray piping assembly

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Figure 2-27
Interface between core spray piping and spargers

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Figure 2-28
Core spray piping supports configuration and details

2.4.2 Safety Assessment*

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2.4.3 Conclusions and Actions

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2.5 Core Spray Sparger

2.5.1 Hardware Evaluation

Component Description and Function

The core spray piping internal to the RPV (Section 2.4) provides the flow path of core cooling water from the vessel nozzle, through the shroud and to the four core spray spargers above the core region. The core spray spargers provide a uniform distribution of the core spray flow over all fuel bundles to assure long-term core cooling when the core cannot be reflooded. On plants with both high pressure and low pressure core spray, each has an independent pair of spargers.

Failure Locations and Product Line Variations

The combined core spray piping and sparger assembly is shown in Figure 2-25. Figure 2-29 shows a typical core spray sparger and the location of welds and attachments subject to failure. Table 2-6 indicates the potential failure locations and product line applicability.

Table 2-6
Potential core spray sparger weld failures

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Figure 2-29
Core spray sparger

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Figure 2-30
Core spray sparger sectional details

2.5.2 Safety Assessment*

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2.5.3 Conclusions and Actions

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2.6 Jet Pump Assembly

2.6.1 Hardware Evaluation

Component Description and Function

The jet pumps are located in the annulus region between the core shroud and the vessel wall and provide core flow to control reactor power. During normal operation, each pair of jet pumps is driven by flow from a single riser pipe internal to the RPV. The jet pump flow induces the reactor core flow from the annulus to the lower plenum. Between 12 and 24 jet pumps are included in

BWR/3 through BWR/6 plants, depending on the plant rating. BWR/2s do not have jet pumps, so this section does not apply.

Each jet pump assembly is composed of two jet pumps with a common riser assembly. Each jet pump consists of an inlet-mixer, and a diffuser assembly. The inlet-mixer assembly consists of a 180 degree elbow, a nozzle section with suction inlets, and a mixing section throat. The inlet-mixer assembly is clamped to the riser transition piece by the beam-bolt assembly, and fits into a slip joint at the top of the diffuser. A riser brace provides lateral support for the riser pipe that supplies drive flow to the two jet pumps.

The inlet-mixer assembly can be removed after detensioning the beam bolt assembly. The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end which is welded in to the shroud support plate. Instrumentation monitors jet pump flow through the diffuser to ascertain their individual and collective flow rates under varying operating conditions.

For post-accident core flooding, the jet pump assures flooding to no less than two thirds of the core height. There is no recirculation line break which can prevent reflooding the core to the level of the jet pump suction inlets.

Jet pump instrument sensing lines provide indication of jet pump flows during normal operation and provide a variable leg for post accident fuel zone water level monitoring.

Failure Locations and Product Line Variations

BWR/3 and BWR/4 inlet-mixers have a single hole nozzle, while BWR/5 and BWR/6 inlet-mixers have multiple 5 hole nozzles. Otherwise, the jet pump designs are similar, with dimensional differences (nozzle I.D., throat I.D., diffuser length, inlet-mixer length, etc.) and different flow capacities for different plant sizes. Table 2-7 identifies potential failure locations.

Table 2-7
Potential jet pump failure locations

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Figure 2-31
Jet Pump Assembly

2.6.2 Safety Assessment*

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Location 1 – Riser Brace to Vessel

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Location 2 – Jet Pump Holddown Beam and Bolt

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Location 3 – Jet Pump Riser Pipe Welds

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Location 4 – Riser Transition Piece Holddown

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Location 5 – Inlet-Mixer Assembly Fabrication Welds

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Figure 2-32
Jet pump riser and brace

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Figure 2-33
Jet pump assembly details (cont'd)

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Figure 2-34
Jet pump assembly details (cont'd)

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Figure 2-35
Jet Pump Assembly Details (cont'd)

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Figure 2-36
Jet Pump Assembly Details (cont'd)

Location 6 – Restrainer Bracket (Part of Riser Assembly)

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Location 7 – Restrainer Bracket to Riser Pipe Weld

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Location 8 – Wedge Assembly and Retaining Bolt (Part of the Inlet-Mixer Assembly for BWR 4, 5, and 6)

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Location 9 -Restrainer Bracket Adjusting Screws

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Location 10 – Diffuser Collar to Diffuser Shell Weld

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Location 11 – Diffuser, Diffuser to Adapter, Adapter to Shroud Support Welds

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Location 12 – Jet Pump Sensing Line and Support Bracket

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2.6.3 Conclusions and Actions

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2.7 LPCI Coupling

2.7.1 Hardware Evaluation

Component Description and Function

Several of the newer BWR/4 plants, as well as, all BWR/5 and BWR/6 plants, have couplings which provide low pressure core injection (LPCI) flow paths from the LPCI vessel nozzles to the inside of the shroud. A flow diverter, installed on BWR/6 plants inside the shroud prevents flow-induced vibration of in-core instrumentation.

The diverter consists of a splash plate welded inside the shroud by four legs. The diverters were added because an optional return path during shutdown cooling uses the LPCI line. The newer BWR/4 and BWR/5 plants contain a baffle arrangement which serves a similar purpose to direct flow downward over the top guide.

Failure Locations and Product Line Variations

LPCI couplings are essentially identical on the BWR/4 and BWR/5 plants. The piping arrangement on BWR/4 and BWR/5 plants directs flow just above the top guide; on BWR/6 plants the injection point is directed to a location below the top guide above 2/3 core height. Table 2-8 shows potential failure locations.

Table 2-8
Potential LPCI coupling failures

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2.7.2 Safety Assessment*

Location 1 – Elbow Welds

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Locations 2 and 3 – Shroud to Thermal Sleeve or Cover Plate to Coupling and Shroud
Location 6b – LPCI Baffle

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Locations 4 and 5 – Strut to Elbow or Strut to Shroud

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Location 6a – LPCI Flow Diverter

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Location 7 – LPCI Coupling Clamp Bolts

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Location 8 – Sleeve Flange to Sleeve and Shroud Welds

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Location 9 – LPCI Coupling Lift Lug Welds

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Figure 2-37
LPCI coupling BWR/6

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Figure 2-38
LPCI flow diverter BWR/6

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Figure 2-39
LPCI coupling assembly

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Figure 2-40
Interface between LPCI coupling and baffle BWR/4-5

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Figure 2-41
Interface between LPCI coupling and baffle

2.7.3 Conclusions and Actions

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2.8 In-core Housing and Dry Tube

2.8.1 Hardware Evaluation

Component Description and Function

The Neutron Monitoring System (NMS) measures neutron flux, which is used to control operation of the reactor and, in some cases, induce a scram signal that would terminate reactor operation. The in-core housings (Figure 2-42) provide a pathway for the NMS detectors to have access inside the reactor core. The detectors are inserted through tubes inside of the in-core housings. The tops of the detectors are spring fitted (Figure 2-44) into the top guide to prevent lateral movement. The in-core housings, however, extend to just above the core plate. The in-core housings from the vessel penetration to the flange outside the vessel are part of the reactor vessel pressure boundary.

The in-core stabilizer hardware provides lateral support for the tubes to prevent movement caused by flow-induced vibrations.

Failure Locations and Product Line Variations

All BWR product lines are similar in the design of the in-core assemblies. Numbers of detectors vary according to plant size and product line. Table 2-9 shows the locations of potential failure.

Table 2-9
Potential in-core failure locations

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2.8.2 Safety Assessment*

Consequence of Failure

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Location 1 – In-vessel ICM Housing Tube Welds

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Location 2 – Vessel Penetration Weld

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Location 3 – In-core Housing Tube to Stabilizer Tack Weld

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Location 4,5 – Tie Bar to Clamp Fillet Weld and Bolt

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Figure 2-42
In-core housing

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Figure 2-43
In-core housing support bars

Location 6 – Dry Tubes

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Figure 2-44
In-core flux monitor dry tube location 6

2.8.3 Conclusions and Actions

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2.9 Orificed Fuel Support

2.9.1 Hardware Evaluation

Component Description and Function

The Orificed Fuel Support (OFS) casting supports the weight of the fuel assemblies and distributes core flow into the fuel bundles. Two types of OFS castings are used in BWR product lines.

The Four-Lobed OFS (Figure 2-45) is a cylindrical structure with four internal compartments and a central opening for the positioning of the control rod blade. The fuel bundles are inserted into each of the four internal compartments. The OFS casting provides lateral support and alignment to the bottom of the fuel bundles. The weight of the fuel bundle is transferred through the OFS casting to the control rod guide tubes. The coolant flow into the fuel assemblies is regulated by an orifice located on the side of the lower portion of the OFS casting. Orifice sizing varies on the different product lines.

The peripheral fuel support consists of a single opening cylindrical structure that provides lateral support to only a single peripheral fuel assembly. The flow regulating orifice on the peripheral fuel support is located directly below the top opening.

Failure Locations and Product Line Variations

The OFS is a cast piece which would only result in a single failure location. The only product line variations consist of variations in the inlet orifice sizing. The peripheral fuel support has its only weld to the core plate, and is therefore discussed with the core plate in Section 2.3.

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Figure 2-45
Orificed fuel support

2.9.2 Safety Assessment*

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2.9.3 Conclusions and Actions

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2.10 Shroud

2.10.1 Hardware Evaluation

Component Description and Function

The shroud was described in BWROG submittals provided to the NRC in 1994 [5, 6].

Failure Locations and Product Line Variations

Potentially significant failure locations were described in BWROG submittals to the NRC made in 1994 [5, 6].

2.10.2 Safety Assessment

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2.10.3 Conclusions and Actions

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2.11 Shroud Support

2.11.1 Hardware Evaluation

Component Description and Function

The shroud support provides the structural connection between the shroud and reactor vessel wall and isolates the shroud annulus from the lower plenum. The shroud support structure bears the weight of the jet pumps, shroud, core plate, top guide, shroud head and separators (~ 200 kips). During normal operation, the shroud support also must react to an upward load (~ 1,000 kips) resulting from the differential pressure which exists between the lower plenum and top of the shroud head. During normal operation, there is a net upward load on the shroud support.

In a recirculation line LOCA, the shroud support forms part of the core coolant envelope which is needed to maintain the two-thirds height water level core coverage in BWR/3-6 plants.

Failure Locations and Product Line Variations

Three types of shroud support are in use in BWRs: (1) BWR/2 designs utilize a skirt type support (Figure 2-46); (2) Pedestal type supports (Figure 2-47) are used in the majority of BWRs; (3) plate and gusset type supports (Figure 2-48) are used in BWR vessels provided by Combustion Engineering (CE), except in one CE-built BWR/4 (Figure 2-49) where the support is a thick alloy steel plate. Table 2-10 correlates shroud support type, figure, and failure location. Table 2-11 correlates shroud support type with reactor vessel.

Table 2-10
Shroud support joint product line variations

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Table 2-10
Shroud support joint product line variations (continued)

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Table 2-11
Reactor vessel shroud support configuration

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Figure 2-46
BWR/2 skirt shroud support

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Figure 2-47
Pedestal shroud support

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Figure 2-48
Gusset type shroud support type 3

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Figure 2-49
Shroud support plate only

2.11.2 Safety Assessment*

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Skirt Type Shroud Support (BWR-2)

Location 1 – Shroud Support Ring to Skirt Weld (H8)

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Location 2 – Skirt to Vessel Wall Weld (H9)

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Location 3 – Fabrication Welds in Cone Skirt

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Location 4 – Baffle Skirt(Diffuser)

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Pedestal Type Shroud Support

Locations 1, 2, 3, 4, 7 (H8, H9) – Shroud Support Plate and Pedestal Welds

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Location 5 – Radial Seam Welds

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Location 6 – Support Cylinder Vertical Seam Welds

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Location 8 – Cylinder to Shroud Buildup

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Location 9 – Stiffener to Pedestal

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Plate and Gusset Type Shroud Support

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Location 6 – Radial Seam Welds and Support Cylinder Seam Welds

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2.11.3 Conclusions and Actions

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2.12 Shroud Support Access Hole Cover

2.12.1 Hardware Evaluation

Component Description and Function

Access hole covers (AHCs) are located in the shroud support plate of BWR jet pump plants. Each shroud support plate in the vessel has two access hole covers located approximately 180 degrees apart, except some BWR/6s, which have only one. These access holes were used for access during construction and were subsequently closed by welding a plate in the hole. The sole function of access hole covers in operating plants is to maintain a leak-tight barrier between the annulus and lower plenum.

Failure Locations and Product Line Variations

Figures 2-50 through 2-56 show the access hole cover plate configurations for BWRs [9]. Within the different shroud support configurations, there are several possible access hole cover configurations, consisting of different cover thicknesses and weld joint geometries. As a result, the BWRs are categorized, by similarity, into nine different groups. Table 2-12 presents the nine access hole cover configuration groups.

Table 2-12
Access hole cover configurations

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2.12.2 Safety Assessment*

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2.12.3 Conclusions and Actions

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Figure 2-50
Thin AHC and conventional shroud support plate with ledge

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Figure 2-51
Intermediate AHC, conventional shroud support plate without ledge

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Figure 2-52
Thick AHC and conventional shroud support plate with ledge

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Figure 2-53
Thick oval AHC and conventional shroud support plate with ledge

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Figure 2-54
Thin AHC and thick shroud support plate with ledge

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Figure 2-55
Retrofit design for some BWR/4s and BWR/5s

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Figure 2-56
BWR/6 single AHC design

2.13 Top Guide/Grid

2.13.1 Hardware Evaluation

Component Description and Function

The top guide on BWR/2, /3, /4, and /5 plants (Figure 2-57) consists of a grid of square holes (egg crate), which maintains the alignment of control rods and fuel channels during normal operation, pressure transients and seismic events. During normal operation, there are no significant loads on the top guide.

Top guides maintain alignment and spacing at the top of fuel assemblies. This is a safety-related function because the top guide is part of the core support structure.

The top guides are positioned by four vertical or horizontal aligner pins. Bosses or sockets are welded to both the top guide and the shroud to engage the pins. The typical BWR/2/3/4/5 configuration includes a radial gap of greater than 4 inches between the outside diameter of the top guide and the inner radius of the shroud.

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Figure 2-57
Typical BWR/2 – 5 top guide assembly

The top guide on BWR/6 plants (Figure 2-58) consists of a solid plate which has square holes for each fuel cell machined out, and is bolted in place along with the upper shroud; therefore, the discussion of wedges and the grid above is not applicable to BWR/6 (discussion of Location 17).

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Figure 2-58
Typical BWR/6 top guide (grid) location 17

In the case of the BWR/3 and early BWR/4 top guide designs, the seismic and other dynamic loads from the fuel assemblies are transferred to the shroud solely through the alignment pins.
In BWR/2s, many BWR/4s and all BWR/5s, the top guide is further restrained within the shroud by brackets or wedges which are located around the upper perimeter of the top guide and provide additional load transfer paths between the top guide and the upper shroud.

BWR/2, /3, /4, and /5 plants attach the cover plate to the rim surrounding the grid with numerous pins; the bottom plate is attached in most cases to the rim with welds; in a few cases they are an integral machined piece. The rim, top, and bottom cover plates are fabricated from plate. The rim, top, and bottom cover plates form a ring beam that is the load path for fuel lateral loads via the beam grid.

Although the weight of the top guide exceeds the upward force during normal operation, most BWR/2, 3, 4, and 5s (Table 2-13) have holddown latches or "C" clamps which provide vertical restraint following seismic or design basis events which produce a vertical lift.

Failure Locations and Product Line Variations

Table 2-13 describes the potential locations of failures in BWR/2-6 plants. The variations of components among U.S. BWR plants are summarized in Table 2-14.

Table 2-13
Potential top guide component failures (Figure 2-57)

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Table 2-14
Top guide plant variations

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2.13.2 Safety Assessment*

Location 1 – Grid Beam to Beam Crevice Slot
Location 4 – Grid Beam to Rim Top and Bottom Cover Plate Pins
Location 6- Fuel Guard Welds and Bolting

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Locations 2 and 3 – Aligner Pins and Sockets in Top Guide and Shroud

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Figure 2-59
Grid beam to beam crevice slot location 1 detail B from figure (reference Table 2-13, 2-14)

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Figure 2-60
Aligner pin assemblies locations 2 and 3

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Figure 2-61
Location 4 grid beam to rim top and bottom plate bracket

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Figure 2-62
Location 5 grid beam to rim weld

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Figure 2-63
Wedges and holddowns

Location 5- Grid Beam to Rim Connector Weld

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Location 7 – Top Guide Restraining Wedges

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Locations 8 and 9 – Holddown Assemblies

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Locations 10 and 11 – Rim Pins and Bottom Welds

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Location 12 – Rim and Cover Plate Fabrication Welds

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Location 13 – Eye Bolt Boss
Location 15 – Threaded Boss to Top Ring
Location 16 – Lifting Lug to Rim Bolt or Weld

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Location 14 – Support Brackets to Shroud Welds

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Location 17 – Integral Top Guide Fabrication Welds

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2.13.3 Conclusions and Actions

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Figure 2-64
Top guide details

2.14 Vessel Instrumentation

2.14.1 Hardware Evaluation

Component Description and Function

Instrument nozzles penetrate the reactor vessel wall at various elevations (Figure 2-65) and are welded to the inside of the reactor vessel. Reactor vessel instrumentation utilizes these nozzles to measure vessel water level based on differential pressure between variable leg taps from penetrations in the liquid region of the core shroud annulus and taps in the steam region which provide a constant reference (Figure 2-65). Water level and various other reactor pressure instruments use these nozzles to provide signals for safety functions. Welds in the instrument lines external to the vessel penetration weld are not in the scope of this evaluation.

Failure Locations and Product Line Variations

The number of instrument nozzle penetrations vary with product line. Earlier product lines contain two sets of penetrations per division while later product lines contain four sets. Potential failure locations and product line applicability are summarized in Table 2-15.

Table 2-15
Instrumentation failure locations

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Figure 2-65
Location of water level instrument taps

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Figure 2-66
Instrumentation nozzle configurations

2.14.2 Safety Assessment*

Location 1 – Vessel Wall to Penetration Weld

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Table 2-16
Water level trip functions

Trip Water Level	Function
High-Level Trip (Level 8)	• Trip RCIC and HPCI or HPCS • Close main steam stop valves • Trip reactor feedwater pumps • Reactor Scram (BWR/5, 6)
Scram-Level Trip (Level 3)	• Reactor scram • ADS Permissive • Recirculation flow runback • Close RHR shutdown cooling isolation valves
RCIC Level Trip (Level 2)	• Initiate RCIC and HPCI or HPCS • Start HPCS diesel generators • Close containment isolation valves except RHR shutdown isolation valves and MSIVs on some plants • Recirculation pump trip
ECCS Level Trip (Level 1)*	• Initiate LPCI and core spray • Initiate ADS (with permissives) • Start emergency diesel generators • Close MSIVs on some plants

* BWR/2 and early BWR/3 plants do not have instrumentation which measures levels below Level 2. The above listed Level 1 trips are at Level 2 on these plants.

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2.14.3 Conclusions and Actions

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3
NON-SAFETY-RELATED COMPONENTS

This section discusses components which are categorized as non-safety-related. Failure of these components may cause an operability concern or be a source of loose parts, but otherwise has no safety impact. As such, actions to assure safety function are not applicable and are therefore not provided. This section discusses the functions and consequences associated with failure of the steam dryer, steam separator/shroud head, feedwater sparger and surveillance specimen holder, and provides the basis for their non- safety-related categorization.

3.1 Steam Dryer

3.1.1 Component Description and Function

The steam dryer assembly (Figure 3-1) is mounted in the reactor vessel above the separator assembly and forms the top and sides of the wet steam plenum. During refueling operations, the dryer is removed to allow access to the reactor core. Vertical guide rods on the inside of the vessel provide alignment for the dryer assembly during re-installation (these guide rods also are used for installation of the shroud head (Section 3.2). The dryer assembly is supported by brackets extending inward from the vessel wall and is held down in position during operation by the vessel head holddown brackets.

Steam from the separators flows upward and outward through the dryer's drying vanes. These vanes are attached to a top and bottom supporting member forming a rigid integral unit. Moisture is removed and carried by a system of troughs and drains to the pool surrounding the separators and then into the recirculation downcomer annulus between the core shroud and reactor vessel wall.

The steam dryer's sole function is to remove moisture from steam in order to minimize erosion of piping and the turbine by reducing its moisture content. The dryer is not safety-related.

The steam dryer is restrained by four brackets located approximately 0.5 inches above the steam dryer lifting lugs after the vessel head has been lowered on to the flange. The holddown brackets prevent lift of the dryer caused by transient pressure following a main steamline break. The brackets are welded to the inside of the vessel head. During removal or insertion of the dryer, guide rods ensure that the dryer is aligned with the shroud and does not impact vessel internal components during insertion or withdrawal. The guide rods are held in position at each end by brackets welded to the vessel. Between four to six dryer support brackets are welded to the inside of the vessel to support the dryer assembly weight.

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Figure 3-1
Steam dryer assembly

3.1.2 Failure Consequences

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3.2 Shroud Head and Separators

3.2.1 Component Description and Function

The steam separator assembly (Figure 3-2) consists of a domed base (shroud head) on top of which is welded an array of standpipes with steam separators located at the top of each standpipe. The fixed axial flow type steam separators have no moving parts and are made of stainless steel. In each separator (Figure 3-3), the steam-water mixture rising through the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the water from the steam in each of the stages. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits from the lower end of each stage of the separator through drain tubes and enters the pool that surrounds the standpipes to join the downcomer annulus flow.

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Figure 3-2
Shroud head and separator assembly

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Figure 3-3
Separators

The shroud head assembly rests on the top flange of the core shroud and forms the cover of the core discharge plenum region. The shroud head is bolted to the core shroud flange, by long holddown bolts which, for ease of removal, extend above the separators (BWR/6 has a shorter bolt arrangement). Typically, 36 to 48 bolts (Figure 3-4) are used; however, the exact number of bolts and bolt sizes varies. The objective of the long-bolt design is to provide access to the bolts during reactor refueling operations with minimum underwater tool manipulation during the removal and installation of the assemblies. It is not necessary to engage threads in connecting the shroud head. A tee-bolt engages in shroud lugs and the nut on the top of the shroud head bolt is tightened to only nominal torque. Final loading is established through differential expansion of the bolt and compression sleeve.

Many of the earlier pre-BWR/6 shroud head designs do not have seismic pins to transfer shear loads from the shroud head to the shroud. For these plants, the shroud head bolts and guide pins must carry lateral loads. The shroud head guide pins have been shown capable of carrying lateral seismic loads in analyses performed to demonstrate that no safety issues are involved, even though these members were not originally designed to carry seismic loads.

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Figure 3-4
Shroud head bolt

During installation, the separator base is aligned on the core shroud flange with guide rods and guide pins. The guide rods are supported by brackets mounted to the vessel wall and extend from the RPV flange elevation to the shroud head flange elevation. The guide rods engage brackets in the shroud head during insertion and removal of the head moisture separator assembly during outages. The guide rods ensure alignment of the head with the shroud guide pins and prevent contact between he shroud head and vessel internal components. The sole purpose of the guide rods is to maintain head alignment and prevent head to vessel contact during shroud head insertion and withdrawal. The guide rods have no safety function.

3.2.2 Failure Consequences

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3.3 Feedwater Sparger

3.3.1 Component Description and Function

There are between four to six feedwater spargers in the vessel, depending on individual plant design. Each sparger (Figure 3-5) has a bracket at each end that supports the weight of the sparger and reacts to the loads from the discharge of feedwater into the vessel. The brackets are welded to the vessel wall just above the top of the shroud head. A thermal sleeve separates the nozzle flow from the nozzle inner surface to minimize thermal fatigue to the nozzle. There are several thermal sleeve designs; all conduct the feedwater flow from the safe end to the sparger.

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Figure 3-5
Feedwater sparger details

The feedwater spargers distribute the flow of feedwater to the vessel through small nozzles to provide uniform mixing of water in the annulus and provide margins against jet pump cavitation. The thermal sleeve tee is located in the middle of each sparger segment, and the segment is shaped to fit the contour of the reactor vessel wall.

As indicated in Table 2-5, the feedwater sparger is a component of various core makeup and shutdown cooling systems. However, for those systems sparger distribution is not critical to safety, so the spargers are not safety-related.

3.3.2 Failure Consequences

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3.4 Surveillance Capsule Holder

3.4.1 Component Description and Function

The surveillance capsule holder (Figure 3-6) contains reactor pressure vessel material used to monitor the irradiation embrittlement of the vessel. A bracket holds the surveillance specimen holder to the vessel wall. Specimen holders are removed with the vessel specimens.

3.4.2 Failure Consequences

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Figure 3-6
Surveillance specimen detail

4 CONSIDERATION OF LOOSE PARTS

This section evaluates the potential impact on loose parts generated in the vessel due to cracking of vessel internal components. Additionally, evaluation is included to assess why loose parts do not negatively affect safe shutdown and offsite dose. This analysis is also valid for loose parts generated outside the reactor vessel, as long as they reach the regions inside the vessel considered as source locations, and their sizes are within the range of sizes considered in this evaluation. Operating experience of most plants indicates that loose parts have not significantly affected plant operation. However, there has been some degradation of certain components and/or systems at times. It is important to ensure that programs are in place to effectively eliminate the introduction of loose parts, promptly identify loose parts that enter the vessel, and implement appropriate corrective action upon identification of loose parts wherever they may be. It is a good practice to keep an inventory and history of lost parts that includes size, part identification, the date the part was lost and, if applicable, the date it was recovered. The evaluation is general in nature and does not assess the effects on the basis of individual part geometries or material properties, as these are resolved on a case-by-case basis. The generation of multiple loose parts that could arrange themselves in a manner so as to cause unacceptable conditions is not considered to be credible for the purpose of this discussion.

Only those loose parts, which are not detectable and could impact the safe plant operation and shutdown capability, are generally considered to be of safety significance. Loose parts that are detectable due to their observable collateral impact on plant operation would indicate that an abnormal plant condition exists and the plant would normally be brought to a safe condition by operator action.

Because several plant-specific features can determine the acceptable loose part size limits for safety and operational concerns, only general criteria are provided.

4.1 Safety or Operational Concerns from Postulated Loose Parts

Potential loose parts may represent a safety or operational concern. The following safety concerns are considered in this evaluation:

- Potential for interference with main steam isolation valves,
- Potential for interference with S/RV operation,
- Potential for fuel bundle flow blockage and consequent fuel damage,
- Potential for interference with control rod operation,
- Potential for impact damage on reactor internals,

- Potential for corrosion or adverse chemical reaction with other reactor materials,
- Potential for interference with HPCI and RCIC operation,
- Potential for interference with RWCU or RHR isolation valves,
- Potential for interference with the nuclear instrumentation, and
- Potential for Interference with Operation of the RHR Pumps and Heat Exchangers

The following operational concerns are considered in this evaluation

- Potential for fuel fretting,
- Potential for interference with operation of the RWCU pumps, heat exchangers and filter demineralizers,
- Potential for flow blockage of the reactor vessel bottom head drain, and
- Potential for impairment of recirculation system performance.

Loose parts analyses are distinguished by the following properties:

Migration – A part can migrate within the reactor vessel and this evaluation analyzes the transport of the loose part in the vessel and associated systems. This is used as input to the safety and operational evaluation of a loose part analyses. Transport involves weight, lift, and buoyancy forces on the components. Thus, if the part fell in a region where there is significant flow during reactor operation, the part might be lifted and carried by the flow in that location.

The transport of large loose parts in the reactor vessel is gravity-dominated, due to the relative shape and weight of the loose part. Because the flow clearances between the reactor vessel regions are small, the large parts would likely remain in the regions in which they are generated. It is considered highly unlikely that large parts like the relatively large plate type material from the steam dryer, or a part of a jet pump beam will follow a migration path to the lower plenum and cause fuel bundle flow blockage that would negatively affect safe shutdown and offsite dose.

The transport of small loose parts or debris in the reactor vessel would be gravity-dominated, flow-dominated, or a combination of both. Because the flow clearances between these regions are on the same order of magnitude as the size of the parts, certain small parts or debris could be transported away from the location in which they were generated.

Source Location – This evaluation considers parts that have been lost in or migrated to the upper plenum, steam separator/dryer, downcomer, and lower plenum regions. The possibility of loose parts being carried over by the flow to locations where they could present a safety concern or negatively affect safe shutdown and offsite dose is discussed.

Loose parts of any size may be generated in the following regions:

Upper Plenum

The core flow velocity in the upper plenum is on the order of 10 ft/sec, which is insufficient to lift large loose parts away from their source location. However, it may be sufficient to transport small parts or debris into the steam separators. Loose parts from the upper plenum, which are large enough that gravity dominates over upward flow, would most likely fall into the core region and rest on the top guide. Examples of such loose parts are core spray. The geometry of certain parts and their components may be such that they would not be able to pass through the fuel bundle upper tie plate openings. If a loose part were transported to the core bypass region, it would most likely settle on the core plate and remain there because the maximum velocity at the core support plate is low and not likely to be sufficient to lift the part. During normal operation, flow through the bundle would likely keep small parts that could fit through the upper tie plate opening from entering the bundle. During shutdown, there is the possibility of the small parts falling into the bundle or even larger parts falling in the bypass region during fuel shuffle.

If a loose part were to fall on the core top guide assembly or on a fuel bundle upper tie plate grid, significant blockage of the fuel bundle flow could occur. If several fuel bundles were blocked sufficiently to affect the total core flow, the plant operators would observe a power reduction and the plant would be brought to a safe shutdown. If the extent of blockage can not be observed, fuel damage could occur due to inadequate coolant flow through the affected bundles, but would be detected by off gas monitors and would not be a safety concern or negatively affect safe shutdown and offsite dose. Likewise, possible top guide and fuel bundle wear damage from flow-induced motion of a loose part would not be extensive enough in one fuel cycle to be a safety concern.

Debris generated in the upper plenum would most likely be carried into the steam separators, where it would be carried with the water into the downcomer region. If the debris were light enough to pass through the separators to the dryers, again it would most likely be transported to the downcomer region. Very light debris could pass to the main steamlines, but such particles are not expected to cause any safety consequences.

Steam Separator/Dryer

Transport of parts from the steam separator is highly unlikely. If pieces break loose from the separator and are carried with steam, they would be stopped in the dryer area or would drop into the collecting trough and then pass through a drain tube to the downcomer area (see item below for discussion of parts in the downcomer area). The transport of parts from the dryer could conceivably go up into the steam line. Migration in the steam line may take the part to a Main Steam Isolation Valve (MSIV) or to an opening for a Safety Relief Valve (SRV). It is not expected that the part would stay in the valve area with the MSIV in the open position. MSIVs are redundant and failure of one MSIV to close has been analyzed and will not negatively affect safe shutdown and offsite dose. SRVs are usually mounted on stub tubes and there is no flow in the tube when SRV is closed. It is highly unlikely that the loose part can position itself to enter the SRV opening and be the right size to completely block the opening. There are redundant SRVs and failure of one SRV is not a safety concern and would not negatively affect safe shutdown and offsite dose.

Downcomer

Any size loose parts generated from components in the downcomer or passed there from the upper plenum or separator/dryer would be drawn downward by gravity or flow. The parts would settle on the core shroud support ring, the jet-pump assembly or be drawn to the recirculation pump suction. The maximum size small part that could be transported into the lower plenum by being drawn into the jet pump either directly or through the recirculation pump depends on the plant unique jet pump dimensions and nozzle design.

If blockage of the jet pump assembly or the recirculation pump suction were to occur, recirculation and jet pump flows would be affected and detected by routine operator surveillance. The pump vibration monitor could also detect damage to a recirculation pump impeller. If the effect is detectable, operator action could be expected to bring the plant to a safe condition. Other settling locations would not impact the core flows and therefore would not present a safety concern.

Both small and large parts in the downcomer region could likely be swept into the recirculation suction line where the small parts would pass through the recirculation pump without damage to the pump and would be carried by the jet pump driving flow into the lower plenum. The larger parts, depending on size, can pass through the recirculation pump, and if they do, they can pass through the jet pump nozzle, again depending on size. A large part like a recently analyzed part of a broken jet pump beam caused damage to a recirculation pump impeller. The damage required impeller replacement, but detection did not allow the part to negatively affect safe shutdown and offsite dose. Any debris drawn into the jet pump directly from the downcomer also would enter the lower plenum. Debris entering the recirculation pump suction could migrate to the Reactor Water Cleanup System (RWCU), where it would be removed by the RWCU filters.

Lower Plenum

In the lower plenum, the vertical component of the flow velocity is less than 10 ft/sec and would be insufficient to cause large parts to be lifted by the flow. Consequently, large loose parts which have been enerated from lower plenum components or passed through the jet pump throat in the downcomer annulus, would settle to the bottom head region where they would present no safety concern and would not negatively affect safe shutdown and offsite dose. Furthermore, the radial component of the flow velocity ranges from 7 ft/sec in the periphery to less than 1 ft/sec at the center and would tend to move the parts inward toward the reactor vessel centerline. This inward movement would be restricted by, the "forest" of nuclear instrumentation and control rod guide tubes in the lower plenum.

Geometry – The loose parts can be in a variety of sizes and shapes from weld debris of indeterminate shape and less than an inch long to large components such as relatively large plate type material from the steam dryer. Other examples of large loose parts include a jet pump beam, the core spray nozzles and parts of the core spray sparger assembly, which can be several inches long and several inches wide and possibly several inches thick. This evaluation considers three sizes of loose parts: (1) large parts greater than 2 inches, (2) small parts of about 1-2 inches and (3) debris smaller than about 1 inch in size.

4.1.1 Potential for Interference with Main Steam Isolation Valves

The size of potential loose parts that might migrate into a steam line can be small or large and may be carried through the valve by the high steam velocity. The small parts are normally much smaller than the main steam isolation valve (MSIV) opening and, therefore, cannot block the valve. The larger parts, depending on size, can pass through the valve. There are no crevices in the valve body where the parts can be caught. In the upstream side of the valve, there is a fixed liner leg on vertical centerline, which could become bent or broken by a large part hitting it. The loose part and/or the broken liner leg may prevent the valve from closing. It is expected that the likelihood of blocking one valve and failure of the remaining valve (in the same line) to close is remote. Similarly, the likelihood of both valves to be blocked (multiple loose parts) is remote. The potential loose part would not be expected to interfere with the safety or normal operation of the valves such that isolation would be impaired or safe shutdown and offsite dose be negatively impacted.

Therefore, it is concluded that there is no safety concern associated with the potential for blockage of the MSIVs due to the presence of potential loose parts.

4.1.2 Potential for Interference with SRV Operation

As mentioned above, loose parts within a certain size range can enter the main steam line. Since the SRVs take steam from the main steam lines, the potential for these parts to enter the SRV and interfere with their operation must be addressed. During normal operation SRVs are closed, so there is no flow to draw the loose parts into the valves. SRV activation occurs during an Anticipated Operational Occurrence (AOO) or a Loss of Coolant Accident (LOCA). Impact of a loose part along with LOCA or AOO is not considered in the loose parts analysis process. This is due to the fact that the failure of a system or component due to a loose part, coincident with LOCA or AOO is highly unlikely.

Therefore, the potential loose parts are not expected to interfere with SRV operation and negatively impact safe shutdown and offsite dose.

4.1.3 Potential for Fuel Bundle Flow Blockage and Consequent Fuel Damage

Flow blockage of individual fuel channels can result in local overheating and, consequently, fuel damage to the affected fuel channels. The entire core would not be impacted. Fuel channel blockage depending on the amount, can be detectable during normal operation. However, if fuel damage were to occur, plant operators would have immediate indication from the off gas monitors or the main steamline radiation detectors. Small parts, which have migrated to the lower plenum from the downcomer or generated from lower plenum components could be lifted by the flow and carried to the inlet orifice. For the case where these potential small-loose parts migrate to the lower plenum, the migration to the fuel inlet orifices requires the loose part to remain in a horizontal orientation. Migration in a horizontal position is considered unlikely because of the close proximity of the guide tubes (about 1-6 in.) and the low flow velocities in the vessel bottom to lift the part in between the guide tubes. It is postulated that the potential loose parts may enter through the fuel inlet orifices, which range in size from about

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inches in diameter, depending on location and specific plant design [12]. Partial flow blockage of a fuel inlet orifice can lead to initiation of boiling transition [13] or possibly channel instabilities [14]. Due to the higher lift velocities and smaller orifice sizes in peripheral fuel bundles, small parts are more likely to block fuel channels in peripheral bundles than in central ones. Channel instability is less of a concern in this region due to the lower power distribution and availability of bypass flow.

Smaller parts or debris that are able to pass through the inlet orifice could be stopped in the fuel bundle at the lower tie plate (LTP) or the fuel rod spacers which have smaller clearances than the orifice. The percentage of inlet orifice or lower tie plate blocked by a small loose part will be smaller than that required to initiate boiling transition. There is no safety consequence that can negatively affect safe shutdown and offsite dose from such small blockages.

The transport of large loose parts in the reactor vessel is gravity-dominated, due to the relative shape and weight of the loose part. Because the flow clearances between the reactor vessel regions are small, the large parts would likely remain in the regions in which they are generated. It is considered highly unlikely that large parts like the relatively large plate type material from the steam dryer, or a part of a jet pump beam will follow a migration path to the lower plenum and cause fuel bundle flow blockage.

Therefore, the potential loose parts are not considered to cause a significant safety concern for fuel bundle blockage and consequent fuel damage that would negatively affect safe shutdown and offsite dose.

4.1.4 Potential for Interference with Control Rod Operation

In order for the loose parts to interfere with the control rods, they must migrate into the core bypass region or the guide tubes. When parts migrate to the lower plenum, the direct flow paths from the lower plenum into the bypass and guide tube region are too small for most of the loose parts to pass through. The loose parts might pass through the fuel orifice to reach the lower tie plate. Most of the smaller potential loose parts may not be able to pass through the fuel lower tie plate. The debris filter on the lower tie plates could stop even the smallest parts. If a part passes through a lower tie plate, the spacers will most likely stop it before it can migrate through the bundle and enter the upper plenum region. If the loose part reaches the upper plenum, it can fall in the bypass region and into a control rod guide tube. The upward flow through the steam separators will prevent the parts from migrating through the separators and back into the upper plenum region.

Lighter parts, which have entered the bypass region, would most likely be lifted by the flow. Heavier parts could drop to the top of the core plate in the bypass region or into a fuel support casting, where the part may pass through the gap between the fuel support and the control rod blade to inside the guide tube. Once inside a guide tube, again depending on size, the part may migrate past the velocity limiter to the bottom of the guide tube, where it would most likely stay at the bottom outside edge of the guide tube. If the part moved upward in the guide tube from the outer edge, it would have to migrate to the center area of the guide tube where the control rod drive shaft enters the bottom of the guide tube. In this area, the part would have to orient itself to pass through the narrow gap between the control rod shaft and guide tube, which is highly unlikely. There is no other potential path identified for the loose parts to get to the control rods. In worst case, if a part causes a single failure to scram a control rod, the accident analysis covers this condition.

A large part like a steam dryer plate material or a piece of a jet pump beam may make its way to a valve or pump, but because of its physical size, could not reach the control rods. Loose parts do not, in general, affect control rod drive (CRD) operation because of the torturous path required for a loose part to enter the CRD guide tube. Clearances between the fuel channel and the top guide from the upper plenum are small. While from the lower plenum, access to the CRD guide tube by loose parts is effectively prevented by the integrity of the guide tube.

Therefore, it is concluded that there is a safety concern associated with the potential of a control rod to scram due to the presence of the potential loose parts, but a single failure has been analyzed to not negatively affect safe shutdown and offsite dose.

4.1.5 Potential for Impact Damage on Reactor Internals

It is highly unlikely that most potential loose parts would be carried by steam flow down the steam lines. The large and small size loose part(s) could migrate into the reactor annulus, where they could migrate into the recirculation system and from there into the lower plenum. If a loose part entered a recirculation line, it would have to pass through the jet pump nozzle to enter the lower plenum. The size of the parts would be limited to the ID of the jet pump nozzle(s). The potential loose part can also enter the jet pump through the suction flow from the downcomer region into the jet pump mixer.

If the part enters the lower plenum through the jet pumps, the high downward velocity component will tend to keep it on the bottom of the lower plenum. The factor that will determine whether the part will sweep up off the bottom is dependent on the radial component of the velocity.

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Therefore, fretting wear by loose part(s) is less likely to occur in the lower plenum. Except for the nuclear instrument guide tubes, the components at the bottom of the lower plenum are relatively thick and not expected to wear through. The nuclear instrument guide tubes are vulnerable to fretting, but they do not form a pressure boundary in the lower plenum.

The impingement by these part(s) on the shroud, shroud hardware, recirculation lines, jet pump assemblies, core spray piping, and other reactor internals would normally be of minimal to no consequence because of the relatively small size of the part(s) in comparison to the other reactor internal components. In rare cases, a part may become stuck or trapped in an area like a jet pump nozzle or the jet pump assembly itself and cause wear of the equipment, therefore, hindering its expected performance. Performance deterioration would not be a safety issue and would be detected, if significant, so that corrective action could be taken if necessary. If a hole was fretted by the part in the jet pump nozzle, there is no thermal consequence during normal operation because the water is at the same temperature as the downcomer.

Therefore, the potential loose part(s) are not expected to cause significant damage to the reactor internals and there would be no impact on the ability to maintain the 2/3-core height following a loss-of-coolant accident and would not negatively affect safe shutdown and offsite dose.

4.1.6 Potential for Corrosion or Chemical Reaction with Other Reactor Materials

Because the postulated loose parts are made of materials approved for in-reactor use, there is no safety concern for corrosion or chemical reaction with other reactor materials in the reactor and there is no negative impact for safe shutdown and offsite dose.

4.1.7 Potential for Interference with HPCI or RCIC Operation

As mentioned above, loose parts of a certain size range can enter the main steam line. Since the HPCI and RCIC systems take steam from the main steamlines, the potential for these parts to enter the HPCI and RCIC lines and interfere with system operation must be addressed. During normal operation, both systems are idle, so there is no flow to draw the loose parts into the system piping. If the systems were initiated, the initiation signal would be coincident with or would follow a reactor scram signal. The flow in the main steamline will fall off rapidly following scram, which will allow the potential loose parts to settle out on the bottom of the steamline. In addition, the HPCI and RCIC steam demand is controlled to allow the systems to come up to speed without tripping. This ramping of the steam demand will also provide time for the parts to settle out. However, the HPCI steamline is normally connected to the top of the main steamline and it is unlikely that the HPCI steam flow will be sufficient to draw the part(s) up from the bottom of the main steamline. Similarly, the RCIC steamline is routinely a small pipe connected to the side of the main steamline and it is unlikely that the RCIC steam flow will be sufficient to draw even the smaller parts up from the bottom of the main steamline.

Therefore, the potential loose parts are not expected to interfere with HPCI or RCIC operation and negatively impact safe shutdown and offsite dose.

4.1.8 Potential for Interference with RWCU or RHR Isolation Valves

Some of the potential loose parts could be carried by the RWCU flow to the RWCU or RHR isolation valves. The RWCU isolation valves are redundant isolation valves (inboard and outboard), which are normally open and need to be closed during an accident such as a LOCA. The RWCU system is normally operating and taking suction from the vessel through the RHR and bottom head drain lines. Larger parts are expected to be carried along with the recirculation flow and pass by the RHR connection. The smaller parts are not expected to enter the RWCU piping, other than maybe through the bottom head drain, during normal operation. The RHR system does not operate during normal conditions and there is only RWCU flow through the RHR lines. During normal operation, the flow velocity through the RHR line is expected to be too low to carry a large percentage of the potential loose parts through the RHR piping and into the RWCU piping. This is because of the large diameter of the RHR pipe compared to the RWCU line (for example, 20 inches vs. 6 inches).

The RHR isolation valves are redundant isolation valves (inboard and outboard) closed during normal operation. Because there is no flow in the RHR shutdown cooling line, it is expected that none of the potential loose parts will be drawn into the RHR piping. However, the RHR isolation valves are open with flow through the piping when the plant is in shutdown mode. The isolation valves need to be closed during a drain-down event.

The potential loose parts that may be trapped in a recess area inside the valve body could prevent the isolation valve from performing the complete isolation function as required. The likelihood of blocking one valve and failure of the remaining valve to close is remote. Similarly, the likelihood of both valves to be blocked (multiple loose parts) is remote. The failure scenario is one isolation valve blocked by a part with the other valve failing to close (single failure). Since the potential loose parts that get into the RHR and RWCU systems are usually smaller than these valves, significant obstruction is not likely. In addition, if the part does not lodge at a pipefitting, it most likely will not stay in the valve. Therefore, it would not be expected to interfere with the safety or normal operation of the valves such that redundant valve closure would be impaired.

It is concluded that there is no safety concern for safe shutdown and offsite dose, associated with the potential blockage of the RWCU or RHR isolation valves due to the presence of the potential loose parts.

4.1.9 Potential for Interference with the Nuclear Instrumentation

Some of the potential loose parts may migrate to the lower plenum area and could interfere with the nuclear instrumentation. The part could impinge upon the nuclear instrumentation tubes with the normal flow in the lower plenum. The impingement by the parts on the tubes would be of no consequence because of the relatively small size of the parts. Technical Specifications assure adequate diversity in the neutron monitoring systems. If a loose part caused a failure in a nuclear instrument it would fail downscale and be detected. It would not impact the high neutron flux scram function. LPRM failures are an operational concern, however failure of one string would not limit operation. It is unlikely that multiple pieces, large or small, would cause failures in multiple nuclear instrumentation tubes.

Therefore, there is no safety concern for safe shutdown and offsite dose due to interference of potential loose parts with the nuclear instrumentation.

4.1.10 Potential for Interference with Operation of the RHR Pumps and Heat Exchangers

Smaller loose part(s) could migrate and follow a flow path to the RHR pumps. Larger loose part(s) are not expected to migrate to the RHR pumps. In the event that some of the smaller loose part(s) do enter the RHR line, upon reaching the pump, they would normally pass through without causing any pump damage. Testing the RHR pumps at high flow prior to startup can assure that the part will not be subsequently drawn into the RHR pumps. Upon leaving the pump, the smaller loose part(s) would be discharged into a system pipe leading to the associated heat exchangers, where it would normally not pass through but would cause minimal blockage.

Therefore, there is no safety concern for safe shutdown and offsite dose due to impact of the potential loose parts on the operation of the RHR pumps and heat exchangers.

4.1.11 Potential for Fuel Fretting

Smaller identified loose parts may pass through a fuel lower tie plate and get caught on a spacer, where it could wear a hole in the fuel clad. If there were fuel-cladding leakage, it would be detected by the off-gas system so that appropriate actions could be taken to maintain the off-gas radiation release within acceptable plant specific 10CFR100 limits. All identified Technical Specification actions will be complied with, as required. Additional information and recommended actions are provided in SIL Number 552, "Fuel Failures Caused By Metal Debris".

Therefore, there is some possible operational concern for a smaller loose part to cause fuel fretting, but this does not constitute a safety issue. It may result in an economic or reliability issue, but does not negatively impact safe shutdown and offsite dose.

4.1.12 Potential Interference with Operation of the RWCU Pumps, Heat Exchangers and Filter Demineralizers

Only smaller loose parts could migrate and follow a flow path to interfere with RWCU pumps. As discussed in Section 4.1.7, a loose part is not expected to enter the RWCU piping during normal operation. In addition, it is unlikely that a loose part large enough to damage the RWCU pump would enter the RWCU line. If the potential loose part were discharged into a RWCU system pipe leading to the associated heat exchangers, it would most likely not pass through but would cause minimal blockage. The blockage would be similar to that of leaking tubes that are plugged. If the part passes through the RWCU heat exchanger, it would be trapped at the filter demineralizer and not expected to change demineralizer pressure drop, because of the relatively large demineralizer area in comparison to the loose part area.

Therefore, there is no operational concern for safe shutdown and offsite dose due to the impact of the potential loose parts on the operation of the RWCU pumps, heat exchangers and filter demineralizers.

4.1.13 Potential for Flow Blockage of the Reactor Vessel Bottom Head Drain

Smaller loose part(s), which originated in or migrated to the reactor lower plenum, may enter the bottom head drain line. If an accumulation of smaller part(s) partially blocked the bottom head drain line opening, reactor water conductivity may increase. In addition, the temperature differences between the top and bottom of the reactor vessel and between reactor recirculation loops would need to be monitored so as not to violate Technical Specifications if startup of an idle loop is desired.

Therefore, there is no operational concern for safe shutdown and offsite dose due to the impact of the potential loose parts blocking the bottom head drain. However, there is some possible operational concern for partial flow blockage of the reactor vessel bottom head drain, which may require additional monitoring of conductivity and delta temperature at startup time of an idle recirculation loop.

4.1.14 Potential for Impairment of Recirculation System Performance

Either a single potential loose part or multiple smaller parts from the single loose part could be drawn into the recirculation system. Smaller loose part(s) are expected to pass through the recirculation pump without causing any damage. This is based on past experience where large and heavy items have passed through the pump and the jet pump nozzles without causing damage. The smaller loose part(s) are small and are not expected to cause any detectable flow reduction in jet pump drive flow. Therefore there is no potential for recirculation system performance impairment from the smaller size part(s).

The recirculation pump could pass either a single loose part, or a significant portion of a loose part and sustain only minor damage. The damage may not be indicated by high vibration, but the recirculation system performance could be marginally degraded.

If a single loose part passed through the pump, it is unlikely to lodge in the recirculation discharge valve. If it did, it might affect the safety function of the recirculation discharge valve, which must close in a LOCA to allow ECC injection to be effective (for BWR/3s and most BWR/4s). Since the loose parts are smaller than these valves, significant obstruction is not likely, because if it does not lodge at a pipe fitting, it most likely will not stay in the valve. If a loose part prevents the recirculation discharge valve from fully closing, the loss of LPCI flow will delay the core reflooding, resulting in an increase in the Peak Cladding Temperature (PCT). This would only be an issue for the DBA accident in BWR/3s and most BWR/4s. For smaller breaks, the discharge valve closure function is not required to maintain acceptable PCTs. Based on a bounding evaluation, using realistic models and assumptions with no credit for LPCI or discharge valve closure, one core spray system is sufficient to keep the PCT below 2200°F. Therefore, the loose part would not present a significant safety hazard.

The single loose part might plug a jet pump nozzle. Blockages occur due to objects that are too large to pass through the 180-degree bend at the top of the jet pump and/or through the jet pump nozzles. Blockages have occurred due to objects that fall into the downcomer and pass through the recirculation pump, objects from the valves in the recirculation loop and objects that are flushed into the recirculation loop from connected piping such as the RHR System. The maximum size part is unlikely to lodge in the external recirculation piping, as it's most likely to lodge in the jet nozzle or rams head. If it causes fretting wear thru-wall on these components, there would be a drop in indicated jet pump flow. As stated above, the resulting reduction in ECCS flow would not cause PCT to exceed 2200°F.

For a given recirculation pump speed, a blockage will decrease the flow of the affected jet pump, decrease the flow of the recirculation pump, decrease core flow and thereby decrease thermal power. This is an operating concern with achieving rated core flow. With the exception of the jet pump flow rate; most of these changes are very small so that detectablity is limited. By far, the best indication of an obstruction is a change in the jet pump flow rate in relation to the other jet pumps in the loop (e.g., the flow rate of the jet pump vs. the average flow of the jet pumps in the loop). A deviation of more than 10% flow or 20% differential pressure from the historical data will be an indication of a problem, based on operating experience from several BWRs that have experienced jet pump nozzle obstruction. Depending on the Technical Specifications, the data may not be needed to satisfy the surveillance requirements but the data should be recorded and available for review.

Increased monitoring of the recirculation system, would assure that either damage to the pump, or flow reduction due to the single loose part in the discharge valve could be noted and appropriate steps taken to safeguard the plant.

Therefore, a potential loose part(s) in the recirculation system would not result in an operational or safety concern that would negatively affect safe shutdown or increase the offsite dose.

4.1.15 Examples of Small Loose Part Evaluations

Metallic loose part evaluations performed for operating plants have demonstrated that many loose parts have no adverse safety consequences. Table 4-1 lists small loose part evaluations that illustrate the spectrum of loose parts which been successfully evaluated and have shown no adverse safety consequences.

Table 4-1
Small loose part evaluation examples

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4.1.16 Examples of Large Loose Part Evaluations

Metallic loose part evaluations performed for operating plants have demonstrated that many loose parts have shown no adverse safety consequences. Table 4-2 lists large loose part evaluations that illustrate the spectrum of loose parts, which have been evaluated and shown no adverse safety consequence and have not negatively affected safe shutdown and offsite dose.

Table 4-2
Large loose part evaluation examples

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4.2 Conclusions

Based on general considerations, with redundancy and monitoring, this evaluation shows that safe reactor operation and safe shutdown capability are not compromised for most categories of postulated loose part sizes. Larger and heavier parts like pieces of a jet pump beam or plate from a steam dryer have not negatively affected safe shutdown or offsite dose. While it is possible for loose parts, with particular size and shape, to compromise fuel performance, it is extremely unlikely that such loose parts would result from the failure of internal components. If it were to occur, damage would likely be limited to a single fuel bundle, which would be detected by routine monitoring of the offgas monitors or LPRMs. Therefore, there is no significant safety concern from potential loose parts on fuel. There also is no safety concern for interference with MSIVs, control rod operation, damage to reactor internals, corrosion or chemical reaction with other reactor materials, interference with HPCI or RCIC operation, RWCU or RHR isolation valves, Nuclear Instrumentation and RHR pumps and heat exchangers. There could be some possible operating concerns from the potential loose part(s) with regard to fuel fretting, bottom head drain plugging and recirculation system performance, but none of these would negatively affect safe shutdown or increase offsite dose.

4.3 NRC Safety Evaluation (SE) Limitations and Conditions

In accordance with the NRC Safety Evaluation, when referencing the revised Section 4.0 of this report in licensing applications the following limitations and conditions apply:

1. The information provided in this section is applicable only to BWR/2 through 6 units.

2. Because the information in this report is general in nature and does not assess the effects on the basis of individual geometries or material properties, as those are resolved on a case-by-case basis, and because several plant specific features can determine the acceptable loose part size limits for safety and operational concerns, this report and the associated SE for this revised Section 4 provide only general criteria for acceptability of loose parts in operating BWR/2-6 plants. A plant specific safety assessment is, therefore, required to be performed by a licensee in the event loose parts are detected in its plant.

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14. "Density-Wave Instabilities in Boiling Water Reactors," NUREG/CR-6003, ORNL/TM-12130, September 1992.

The following BWRVIP documents relate to reactor internals issues, but are not directly referenced in this report.

15. "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," GENE-523-113-0894, September 1994. (Transmitted on September 2, 1994 to Donald Brinkman (USNRC) by Carl Terry, Executive Chairman, BWRVIP Assessment Committee. *

16. "BWR Core Shroud Distributed Ligament Length Computer Program," GE-NE-523-113-0894, Supplement 1, September 1994.

17. "Core Shroud Blowdown Load Calculation During Recirculation Suction Line Break by TRACG Analysis for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2," GE-NE-L12,00819-05, September 1994. (GE Proprietary)

18. "Evaluation of Corrosion Assisted Cracking of BWR Vessel Attachment Welds," Interim Technical Report, November 1994.

19. "Calculation of Shroud Loads During a Recirculation Line Break," Report SLI-4-029, November 1, 1994.

20. "BWR Core Shroud Evaluation Load Definition Guideline." Report SL-4942, Revision 0, November 11, 1994.

21. "Main Steam Line Break Analysis with TRACG Model," GENE-523-A198-1294, December 28, 1994.

22. "Request for Information Regarding the Impact of BWR Core Plate and Top Guide Ring Cracking," (transmitted on January 3, 1995 as a letter to Jack Strosnider (USNRC) by Carl Terry, Executive Chairman, BWRVIP Assessment Committee). *

23. "Calculation of BWR/4 Shroud Head Loads and Axial Lift During a Main Steam Line Break," Final Report, SLI-94-031, April 25, 1995.

24. "Core Shroud Repair Design Criteria," Revision 0, August 10, 1994. (Transmitted to USNRC on August 18, 1994 by Bruce McLeod, Technical Chairman, BWRVIP Repair Committee). *

* Documents provided to USNRC.

A
NRC FINAL SAFETY EVALUATION ON BWRVIP-06

98-397A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FILE COPY

September 15, 1998

Carl Terry, BWRVIP Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: SAFETY EVALUATION OF THE BWR VESSEL AND INTERNALS PROJECT BWRVIP-06 REPORT (TAC NO. M93926)

Dear Mr. Terry,

By letter dated October 5, 1995, as supplemented by letters dated December 20, 1996, and June 16, 1997, you submitted the Electric Power Research Institute (EPRI) Proprietary Report TR-105707, October 1995, "BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals (BWRVIP-06)," for NRC staff review and approval. The BWRVIP-06 report provided a generic safety assessment of BWR/2-6 reactor internals to determine the short and long term actions required to assure safe operation with the potential for component cracking. The assessment considered the reactor internal components function during normal, transient, seismic, and DBA conditions. The results of this assessment were intended to provide utilities with a generic reactor internals management plan which could be tailored to meet the needs of the individual utilities. Additionally, this assessment was intended to provide the NRC with information needed to evaluate future cracking in BWR internal components.

The NRC staff has reviewed the BWRVIP-06 report of operation with intergranular stress corrosion cracking (IGSCC) degraded internals and reviewed the results of the "Quantitative Safety Assessment of BWR Reactor Internals (BWRVIP-09)" report with respect to Draft Regulatory Guide (DG) 1061 (now Regulatory Guide (RG) 1.174) and finds, in the enclosed Safety Evaluation (SE), that: (1) if, in some cases, new procedures and/or changes to procedures are needed to detect potential failures, these items need to be considered to be near-term action items, to be implemented as soon as possible; and, (2) common mode failure of reactor internals due to IGSCC was not evaluated in the BWRVIP-06 report and the BWRVIP did not provide an adequate basis for the assumption that only a single failure should be considered.

However, assuming single component failures, the BWRVIP-06 report adequately developed an overall prioritized list for determining which components required the immediate development of inspection and evaluation guidelines. Since the issuance of the BWRVIP-06 report, the industry has implemented effective inspection and repair programs to reduce the likelihood of component failure as a result of IGSCC. The results of these inspections were documented in letters dated April 18, 1997, and March 31, 1998. Although the consequences of multiple failures were not addressed in the report, inspections, evaluation and repairs of components to date, are considered sufficient to provide the staff with a high degree of confidence that multiple, undetected component failures are unlikely and to ensure component integrity for the components evaluated in the BWRVIP-06 report. In addition, this SER finds that the BWRVIP-06 report is acceptable for ranking internal components for the development of inspection and flaw evaluation guidelines.

The NRC's Office of Nuclear Regulatory Research (RES) is presently performing confirmatory research into the possible consequences of multiple, cascading component failures. Should the results from this confirmatory research program identify any significant issues, they will be addressed separately with you.

Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely,

Gus C. Lainas, Acting Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

U S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
SAFETY EVALUATION OF EPRI REPORT TR-105707, OCTOBER 1996, "BWR VESSEL AND INTERNALS PROJECT, SAFETY ASSESSMENT OF BWR REACTOR INTERNALS (BWRVIP-06)"

1.0 INTRODUCTION

By letter dated October 5, 1995, the BWR Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) Proprietary Report TR-105707, October 1995, "BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals (BWRVIP-06)," (Reference 1) for NRC staff review and approval. The BWRVIP-06 report was supplemented by letters dated December 20, 1996, (Reference 2) and June 16, 1997 (Reference 3).

The BWRVIP-06 report provides a generic safety assessment of BWR/2-6 reactor internals to determine the short and long term actions required to assure safe operation with the potential for component cracking. The assessment considers the reactor internal components function during normal, transient, seismic, and design basis accident (DBA) conditions. The results of the BWRVIP-06 report was intended to provide utilities with a generic reactor internals management plan which can be tailored to meet the needs of the individual utilities. Additionally, the BWRVIP-06 report was intended to provide the NRC with information needed to evaluate future cracking in BWR internal components.

Increased occurrence of identified intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) internals prompted the US BWR executives to form the BWR Vessel and Internals Project (BWRVIP) in June 1994, to address integrity issues arising from service-related degradation of these important components. It is apparent to the BWRVIP and the NRC staff that as inspection techniques improve and as more inspections are performed, additional IGSCC related cracking in welded and bolted locations of reactor internals will be identified. On this basis, the BWRVIP submitted this document as a means of exchanging information with the NRC for the purpose of supporting generic regulatory efforts related to assessing the safety consequences of potential cracking of BWR/2-6 reactor internals. The BWRVIP-06 report generically evaluates postulated failures caused by IGSCC in welded and bolted locations of vessel internal components and establishes long-term actions which the BWRVIP stated are appropriate to ensure continued safe operation.

2.0 SUMMARY OF BWRVIP-06 REPORT

2.1 Safety Assessment

The objectives of the BWRVIP-06 report were to perform a qualitative safety assessment of BWR reactor internals and attachments to assure continuing safe operation of BWRs with assumed single component failures, to define short-term and long-term actions needed to ensure safe operation, and to develop an overall prioritized list of components. The staff notes that the prioritized list was used as a guide by the BWRVIP to establish which components require the development of inspection and evaluation guidelines first. The safety assessment

ATTACHMENT

evaluated the consequences of fully failed component locations of safety-related and non-safety-related components. The components addressed in BWRVIP-06 are as follows and are not listed in order of priority:

Safety-related

Control Rod Guide Tube, Control Rod Drive (CRD) Housing and Stub Tube
~~Core Plate dP/Standby Liquid Control System (SLCS) Line~~
Core Plate
Core Spray Piping
Core Spray Sparger
Jet Pump Assembly
Low Pressure Coolant Injection (LPCI) Coupling
Incore Housing and Dry Tube
Orificed Fuel Support
Core Shroud
Core Shroud Support
Access Hole Cover
Top Guide/Grid
Vessel Instrumentation

Non-safety-related

Steam Dryer
Core Shroud Head and Separators
Feedwater Spargers
Surveillance Capsule Holder

Given these components, the BWRVIP-06 report identified the welded and bolted locations of each component where IGSCC would be likely to occur. The BWRVIP-06 report also identifies the different bolted and welded locations for the same component for the several BWR product lines, when applicable. As an example, the core spray piping in a BWR/2 has a different layout and configuration than the core spray piping of BWR/3-6. However, the BWRVIP-06 report does not account for any plant-specific modifications that may have been made to the components during construction or operation.

The BWRVIP considers the BWRVIP-06 report to be a bounding assessment of postulated failures of BWR reactor internals. The worst case assumption of a complete failure at each location of the components was assumed in the BWRVIP-06 report. The safety consequences of the complete failure were evaluated at each location. The staff notes that one complete failure was assumed while all other components susceptible to IGSCC were assumed to be intact (i.e., neither common mode failure nor consequential failures of multiple components due to IGSCC were considered in the BWRVIP-06 report). In some cases, failure of multiple welds of a particular component was considered. However, these cases were generally considered to be non-credible events. The staff further notes that no other consequential failures, such as those due to jet impingement and pipe whip, were considered in the BWRVIP-06 report.

The postulated failures were evaluated for normal operations and design basis accidents (DBAs) such as main steam line break, recirculation line break, and safe shutdown earthquake. The BWRVIP-06 report stated that these evaluated failures are beyond the current design or licensing basis of operating BWRs and that no new design bases are implied by the failures considered in the BWRVIP-06 report. It is important to note that the BWRVIP-06 report's acceptance criterion for the consequence analysis was to achieve a safe shutdown, not to maintain original design margins or maintain long-term core cooling.

Examples of the deterministic evaluation of some safety-related internals follow:

1. Standby Liquid Control System (SLCS) / Core dP: The BWRVIP-06 report concluded that there were no safety consequences due to failure of any portion of SLCS as long as the Emergency Procedures Guidelines (EPGs) for an Anticipated Transient Without Scram (ATWS) are followed.

2. Core Plate: A seismic event in combination with aligner pin weld failures for plants without restraining wedges could result in limited horizontal movement of the core plate. Limited movement below the amount specified in the BWRVIP-06 report would not result in slower scram times; however, movement above the amount specified could result in failure to insert control rods. In this case, SLCS is available to shutdown the reactor.

3. Core Spray Piping: The BWRVIP-06 report stated that, since the core spray piping is inspected every refueling outage, there are no short-term actions required.

4. Core Spray Sparger: Failure of these welds is not detectable during normal operations; however, the spargers are inspected each refueling outage.

5. Jet Pump Assembly: Failure of jet pump welds which cause separation of the jet pump are detectable during normal operations; however, jet pump disassembly during a recirculation line break could result in safety consequences.

Based on the results of the deterministic BWRVIP-06 report, the BWRVIP concluded that no short term actions, beyond planned inspections and possible implementation of new monitoring procedures, were required. The BWRVIP-06 report states that all BWR product lines have sufficient level of safety based on the following:

1. Detectability of component failure by online instrumentation

   The BWRVIP-06 report stated that "..if a location failure which could interfere with safe shutdown during an accident scenario can be detected during plant operation, a safe shutdown can be achieved before the component is challenged by the accident scenario." The BWRVIP-06 report also noted that in some cases, potential detection may require implementation of new procedures to monitor reactor operating conditions.

2. Structural and/or functional redundancy

   In this case, the BWRVIP-06 report assumed that "...if failure of a component location results in the loads being redistributed to other components or locations on the same component which have adequate margin to accommodate the additional loads, there will be no adverse impact on the ability of that component to function in achieving safe shutdown."

3. Detectability of component failure by current inspections

   The BWRVIP stated that "...if inspection is being performed with sufficient frequency and detail, the possibility of significant undetected cracking can be excluded when considering the short-term significance of potential cracking."

4. Low probability of challenging event

   In some cases, postulated failures and a DBA are required to pose any safety consequences. Since DBAs are low frequency events, no short-term actions are required.

The staff notes that all long-term actions generally consist of development of inspection and evaluation guidelines and repair or replacement criteria for the specific components. This is based on the underlying assumption of the BWRVIP-06 report that the developed inspection and evaluation guidelines will assist utilities in identifying potential cracking locations and therefore, fixing the cracked location before the component fails.

2.2 Priority List

The NRC staff met with members of the BWRVIP on April 29 and 30, 1997. During the meeting the BWRVIP presented the current prioritized list of reactor internals. The list is as follows:

<u>High Priority Components</u>

Shroud
Core Spray Piping and Sparger
Shroud Support
Top Guide
Core Plate
Standby Liquid Control System

<u>Medium Priority Components</u>

Jet Pump Assembly

<u>Low Priority Components</u>

Control Rod Drive Guide Tube
Control Rod Drive Stub Tube
Incore Housing
Dry Tube
Instrument Penetrations
Vessel Inside Diameter Brackets
Low Pressure Coolant Injection Coupling

The BWRVIP's intent of the prioritized list is to address the high priority components first by developing inspection and evaluation guidelines and repair or replacement criteria. The staff notes that the list has changed slightly since its development in 1994. These changes were based on inspection findings and resulted in a few components moving to a higher priority on the list. The staff also notes that most of the components on the prioritized list do not have regulatory requirements for inspection. However, some components are inspected every refueling outage or every other refueling outage. For example, NRC Bulletin 80-13 (Reference 4), "Cracking in Core Spray Spargers," requested that utilities perform a visual inspection of the Core Spray Spargers and the segment of piping between the inlet nozzle and the vessel shroud every refueling outage. On the basis of this list, the BWRVIP has and will continue to develop inspection and evaluation guidelines and replacement and repair criteria for the high and medium priority components.

2.3 Quantitative Safety Assessment of BWR Reactor Internals (BWRVIP-09)

The BWRVIP performed a probabilistic risk assessment (PRA) of failure of internal components due to IGSCC. This proprietary PRA was submitted to staff by letter dated June 16, 1997, and is referred to as BWRVIP-09. The purpose of the PRA was to provide additional confidence in the conclusions of the BWRVIP-06 report prioritized list. The PRA evaluated eight components whose failure and occurrence of a low probability event could result in increased core damage frequency. The eight components included the control rod guide tube/housing, core plate, core spray piping, core spray sparger, jet pump assembly, low pressure coolant injection (LPCI) coupling, shroud support access hole covers, and top guide. Access hole covers are located in the shroud support plate of BWR jet pump plants, approximately 180 degrees apart, and are used to cover the access holes used during construction. The staff notes that the analysis evaluated the failure of each of the eight components in conjunction with a loss-of-coolant accident (LOCA) and seismic events. Based on this study, the BWRVIP concluded the following:

1. All BWR product lines possess a sufficient level of safety based on detection of component failure, structural redundancy and low probability.

2. Core damage frequencies are below levels of concern.

3. Analysis supports BWRVIP work prioritization.

The staff reviewed the results of the BWRVIP-09 report but did not evaluate the adequacy of the event tree development or the assigned probabilities. As the basis of the review, the staff used the approach described in Draft Regulatory Guide (DG) 1061 (Reference 5). DG-1061 is intended to improve consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. The principles, process, and approach discussed within DG-1061 also provide useful guidance for the application of risk information to a broader set of activities than plant-specific changes to a plant's current licensing basis, i.e., generic activities. As such, DG-1061 was used to help evaluate the change in core damage frequency as a result of postulated failures of individual components. In implementing risk-informed decision-making, changes are expected to meet the following set of key principles.

1. The proposed change meets the current regulations. This principle applies unless the proposed change is explicitly related to a requested exemption or rule change.

2. Defense-in-depth is maintained.

3. Sufficient safety margins are maintained.

4. Proposed increases in risk, and their cumulative effect, are small and do not cause the NRC Safety Goals to be exceeded.

5. Performance-based implementation and monitoring strategies are proposed that address uncertainties in analysis models and data and provide for timely feedback and corrective action.

Since the PRA was performed for a general set of reactors, the mean core damage frequency of a particular plant to which the BWRVIP-06 report would apply was not known. However, the staff assumed that any particular plant subject to this analysis had a mean core damage frequency less than 1×10^{-4} per reactor year. This assumption was based on the fact that the purpose of the PRA was to verify the prioritized list and conclusions of the BWRVIP-06 report. In addition, during the meeting with the staff on April 29, 1997, the BWRVIP stated that the average core damage frequency from BWR IPEs was 2×10^{-5} per reactor year. Using this assumption, DG-1061 allows for increases in calculated core damage frequency that are very small (e.g., core damage frequencies of less than 1×10^{-6} per reactor year) when combined with the applicable large early release frequency guidelines that are also described in DG-1061.

The staff notes that the BWRVIP started with a bounding assessment approach. In this case, the quantification of models was performed with the probability of IGSCC degraded component failures set to 1. The success criteria for this bounding analysis was that frequencies from all accident sequences were less than 1×10^{-6} per reactor year. If the success criteria was met, no further analysis was necessary. If the bounding case resulted in frequencies greater than 1×10^{-6} per reactor year, then the crack growth model was used to calculate component failure probabilities, the PRA models were re-evaluated, and sensitivity calculations were performed. Based on this procedure, the BWRVIP was able to identify which components are more important to safety based on prevention of core damage and was able to limit all increases of core damage frequency to less than 1×10^{-6} per reactor year. The staff notes that the PRA results verified the priority lists and the conclusions of the deterministic BWRVIP-06 report.

3.0 STAFF EVALUATION

The staff has reviewed the BWRVIP-06 report which states that, in some cases, potential detection of cracking internals may require implementation of new procedures. On the basis of this review, the staff has determined that establishment of any new procedures or changes to procedures for the purpose of detecting potential failures of reactor internals that may be required should be a near-term plant-specific action.

As discussed above, the BWRVIP-06 report evaluated complete failures of one component location while all other components susceptible to IGSCC were assumed to be intact (i.e., neither common mode failure nor consequential failure of multiple components due to IGSCC were considered in the BWRVIP-06 report). The staff has determined that it would have been more appropriate to evaluate the safety consequences of partial cracking in several components during normal operations and DBAs. This is based on the fact that many of the components are subject to the same environmental conditions which are conducive to IGSCC.

The staff notes that this analysis only discusses the capability to achieve a safe shutdown with one component location failed. Long term cooling is not evaluated in this analysis. Acknowledging this fact, the staff continues to have unresolved questions about the BWRVIP's conclusions. For example, the BWRVIP stated that there were no safety consequences due to failure of any portion of SLCS or the core spray spargers. With respect to SLCS, the staff cannot conclude that SLCS will be available if the pipe is crushed and not severed. Additionally, the BWRVIP is recommending reduced inspections of core spray spargers for all BWR/3s and BWR/4s (except Hope Creek and Limerick) which appears to be in conflict with the BWRVIP's conclusions discussed above. The staff has determined that core spray is an important safety system because it provides a uniform distribution of spray to assure cooling when the core cannot be fully reflooded, it helps protect some fuel designs from exceeding their fuel safety limits, and in some cases the high pressure core spray system provides a flow path for injection of boron for SLCS.

The staff is reviewing the SLC issue further in its evaluation of the BWRVIP-27 report. Further, in its review of the BWRVIP-18 report, the staff found the reduced inspection frequency for 304L/316L non-creviced welds acceptable because no cracking of these welds have been reported. However, the staff did not agree that the proposed schedule for non-creviced welds in susceptible material is appropriate since much of the sparger and internal piping that has cracked is non-creviced. This issue is being discussed with the BWRVIP.

However, the industry has implemented effective inspection and repair programs to reduce the likelihood of component failure as a result of IGSCC. Although the consequences of multiple failures have not been specifically addressed in the report, inspections, evaluation and repairs of components to date are sufficient to provide the staff confidence that multiple, undetected component failures are unlikely and to ensure component integrity for the components evaluated in the BWRVIP-06 report. In addition, this SER finds that the BWRVIP-06 report is acceptable for ranking internal components for the development of inspection and flaw evaluation guidelines.

The NRC's Office of Nuclear Regulatory Research (RES) is presently performing confirmatory research into the possible consequences of multiple, cascading component failures. Should the results from this confirmatory research program identify any significant issues, they will be addressed separately with the BWRVIP.

Based on this evaluation, the staff has determined that the BWRVIP-06 report was useful for gaining insights which established a priority list for the evaluation of postulated cracked components. The staff agrees with the BWRVIP that the priority list discussed in Section 2.2 of this SER is reasonable considering the results of the BWRVIP PRA and the staff's engineering judgement.

4.0 CONCLUSIONS

The NRC staff has reviewed the BWRVIP-06 report of operation with IGSCC degraded internals and reviewed the results of the BWRVIP-09 PRA with respect to DG-1061 and finds, in the enclosed Safety Evaluation Report (SER), that: (1) if new procedures and/or changes to procedures are needed to detect potential failures in some cases, these items need to be considered to be near-term plant-specific action items, to be implemented as soon as possible, and (2) common mode failure of reactor internals due to IGSCC should have been evaluated in the BWRVIP-06 report because the BWRVIP did not provide any basis for the assumption that only a single failure should be considered.

However, since the issuance of the report, the industry has implemented effective inspection and repair programs to reduce the likelihood of component failure as a result of IGSCC. Although the consequences of multiple failures have not been specifically addressed in the report, inspections, evaluation and repairs of components to date are sufficient to provide the staff with a high degree of confidence that multiple, undetected component failures are considered unlikely and to ensure component integrity for the components evaluated in the BWRVIP-06 report. In addition, this SER finds that the BWRVIP-06 report is acceptable for ranking internal components for the development of inspection and flaw evaluation guidelines.

The NRC's Office of Nuclear Regulatory Research (RES) is presently performing confirmatory research into the possible consequences of multiple, cascading component failures. Should the results from this confirmatory research program identify any significant issues, they will be addressed separately with the BWRVIP.

5.0 REFERENCES

1. Beckham, J. T. Jr., BWRVIP, to USNRC, "Safety Assessment of BWR Reactor Internals (BWRVIP-06), EPRI TR-105707, October 1995," October 5, 1995.

2. Dyle, Robin, BWRVIP, to USNRC, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-06," December 20, 1996.

3. Terry, Carl, BWRVIP, to USNRC, "BWR Vessel and Internals Project Quantitative Safety Assessment of BWR Reactor Internals (BWRVIP-09)," June 16, 1997.

4. IE Bulletin 80-13, "Cracking in Core Spray Spargers," USNRC, May 12, 1980.

5. Draft Regulatory Guide 1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," USNRC, Office of Nuclear Regulatory Research, March 1997.

B
NRC REQUEST FOR ADDITIONAL INFORMATION ON REVISED SECTION 4.0 OF BWRVIP-06-A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 21, 2006

Bill Eaton, BWRVIP Chairman
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213-8202

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - BWRVIP-06-A: BWR VESSEL AND INTERNALS PROJECT SAFETY ASSESSMENT OF BWR REACTOR INTERNALS

Dear Mr. Eaton:

By letter dated May 11, 2005, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted for the Nuclear Regulatory Commission (NRC) staff review, a revised Section 4.0 "Consideration of Loose Parts" to Electric Power Research Institute Technical Report (TR) 1006598, "BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals (BWRVIP-06-A)." The purpose of this revised section to this report is to provide a general evaluation of the potential impact on loose parts generated in the reactor vessel due to cracking of reactor vessel internal components and to assess how loose parts affect the safe shutdown of the reactor plant and offsite dose. Only Section 4.0 to the BWRVIP-06-A report was revised, which was previously approved by NRC letter dated September 16, 2003.

The staff has determined that additional information is needed to complete the review. The staff's request for additional information (RAI) regarding the revised Section 4.0 to the BWRVIP-06-A report is enclosed. In order to complete the staff's review of the revised Section 4.0 to the BWRVIP-06-A report in an efficient and effective manner, your complete response to the attached RAI is required no later than six months from the date of this letter. If you cannot provide a complete response within six months, please contact John Honcharik at (301) 415-1157 to discuss the withdrawal of the revised Section 4.0 to the BWRVIP-06-A report and its future resubmittal when you are prepared to respond to the RAI. In addition, if you have any other questions regarding the enclosed RAI, please contact Mr. Honcharik.

Sincerely,

Matthew A. Mitchell

Matthew A. Mitchell, Chief
Vessels & Internals Integrity Branch
Division of Component Integrity
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure:
Request for Additional Information

cc: BWRVIP Service List

REQUEST FOR ADDITIONAL INFORMATION
SECTION 4.0 TO BWRVIP-06-A REPORT
BWR VESSEL AND INTERNALS PROJECT
SAFETY ASSESSMENT OF BWR REACTOR INTERNALS

RAI 06-A-1

In Section 4.1 of the BWRVIP-06-A report, the BWRVIP discusses the potential of loose parts resulting from the failure of steam dryers and separators that are carried with the steam. The discussion focused on migration of loose parts to prevent the main steam isolation valve (MSIV) and safety relief valve (SRV) performing their functions. Based on the recent industrial experience with the failure of steam dryers, the staff requests that the BWRVIP address whether large loose parts resulting from steam dryer failure can become a missile or cause damage to other reactor components (for instance, large loose parts entering reactor recirculation system (RRS) suction and causing damage of the recirculation pump).

RAI 06-A-2

In Section 4.1 of the BWRVIP-06-A report, the staff noted that the potential for impact on the standby liquid control system stand pipe is not included. The staff requests that the BWRVIP add this item to the list of concerns.

RAI 06-A-3

In Section 4.1 of the BWRVIP-06-A report, the BWRVIP states that "The geometry of certain parts and their components may be such that they would not be able to pass through the fuel bundle upper tie plate openings." The staff requests that the BWRVIP provide information regarding the size of the upper tie plate opening.

RAI 06-A-4

In Section 4.1.1 of the BWRVIP-06-A report, the BWRVIP discusses potential interference with MSIVs. The BWRVIP indicates that the fixed liner leg on the vertical centerline could be bent or broken when hit by a large loose part that might prevent the valve from closing. It appears that the other valve can also fail to close. The staff requests that the BWRVIP discuss the potential for both valves failing to close due to large loose parts being continuously generated as a result of the steam dryer failure.

RAI 06-A-5

In Section 4.1.2 of the BWRVIP-06-A report, the BWRVIP indicates that potential loose parts are not expected to interfere with the SRV operation because of the following: (1) SRVs are closed during normal plant operation, and as such there is no flow to draw the loose parts into the stand pipe and the valve; and (2) the failure of system components coinciding with a loss of coolant accident (LOCA) is unlikely. Recent industrial experience has suggested that the shear layer which separates the mean flow in the main pipe from the stagnant fluid in the branch can be unstable due to the acoustic resonance in the SRV stand pipe. Therefore, the staff requests that the BWRVIP address the possibility that the loose parts may be drawn into the stand pipe and the valve due to vortex shedding and flow-excited acoustic resonances at the inlet.

RAI 06-A-6

In Section 4.1.2 of the BWRVIP-06-A report, the BWRVIP states, "Impact of a loose part along with loss of coolant accident [LOCA] or anticipated operational occurrence [AOO] is not considered in the loose parts analysis process. This is due to the fact that the failure of a system or component due to a loose part, coincident with LOCA or AOO is highly unlikely." Even though a LOCA is a low probability event, an AOO is an anticipated operational occurrence and hence, the staff believes that safety impact needs to be considered for the failure of a system or component due to a loose part. The staff requests that the BWRVIP address this issue.

RAI 06-A-7

The staff recommends that the BWRVIP address in Section 4.1.3 of the BWRVIP-06-A report the necessary measures that would provide early warnings prior to any occurrence of flow blockage of individual fuel channels due to loose parts. These early warnings would enable the licensees to take proper steps so that local overheating and, consequently, fuel damage to the affected fuel channels can be prevented.

RAI 06-A-8

Regarding Section 4.1.4 of the BWRVIP-06-A report, the staff requests that the BWRVIP provide information regarding the mesh size for the debris filter on the lower tie plate.

RAI 06-9

In Section 4.1.5 of the BWRVIP-06-A report, the BWRVIP discusses the potential for impact damage on reactor vessel internals (RVI) components which focus on the loose parts transporting through the jet pump nozzles to lower plenum area. This section concludes that no significant damage to the RVI components is expected due to the very low core flow in the reactor lower plenum. The staff requests that the BWRVIP discuss the potential damage to the RVI components or instrumentation lines due to loose parts that are carried by the feedwater flow to the reactor and down into the annulus area. This observation is based on the recent industrial experience in Dresden uprated power operation where feedwater probes were found penetrating the feedwater sparger inside the reactor.

RAI 06-A-10

Based on the discussion provided in Section 4.1.6 of the BWRVIP-06-A report, the staff requests that the BWRVIP address possible crevice corrosion due to the presence of a crevice created between a lodged loose part and the RVI component. Crevice corrosion can be more pronounced in RVI components where effective protection due to hydrogen water chemistry and or noble metal chemical addition is not available and the reactor water flow is stagnant.

RAI 06-A-11

In Section 4.1.7 of the BWRVIP-06-A report, the BWRVIP discusses the potential for interference with high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) operation, indicating that during normal operation, both systems are idle and stagnant, and no flow will draw loose parts into the system piping. As requested previously, the staff requests that the BWRVIP address whether loose parts could be drawn into the system piping when the shear layer at the opening to the HPCI and RCIC systems become unstable due to a higher flow of the main steam line because of acoustic resonance.

RAI 06-A-12

The BWRVIP in Section 4.1.11 of the BWRVIP-06-A report specifies Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," limits for off-gas radiation release. However, the staff notes that the main concerns are related to ALARA and hence 10 CFR Part 20, "Standards for Protection Against Radiation," is applicable. The staff requests that the BWRVIP address the application of 10 CFR Part 20 with regards to ALARA considerations. The staff also requests that the BWRVIP submit GE Service Information Letter (SIL) Number 552, "Fuel Failure Caused By Metal Debris."

RAI 06-A-13

The staff notes that some licensees keep a list of the lost parts from the beginning of operation of the plant. The staff recommends that the BWRVIP add a section to the report to indicate that keeping an inventory and history of the lost parts (indicating the number, size, date the part was lost and the date it was recovered) is good practice to keep a track of the lost parts.

RAI 06-A-14

The staff requests that the BWRVIP reference the following document in Section 4.0 of the BWRVIP-06-A report: NRC Regulatory Guide 1.133, Revision 1, "Loose Parts Detection Program for the Primary Systems of Light Water Cooled Reactors," 1981.

C
BWRVIP RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION ON REVISED SECTION 4.0 OF BWRVIP-06-A

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2007-358 BWR Vessel & Internals Project (BWRVIP)

November 30, 2007

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: Holly Cruz

Subject: Project No. 704 - BWRVIP Responses to NRC Request for Additional Information on Section 4 of BWRVIP-06-A

Reference:

1. Letter from Matthew A. Mitchell (NRC) to Bill Eaton (BWRVIP Chairman), "Request for Additional Information – BWRVIP-06-A: BWR Vessel and Internals Project Safety Assessment of BWR Reactor Internals," dated December 21, 2006.
2. BWRVIP letter 2002-138 from Carl Terry (BWRVIP Chairman) to Document Control Desk (NRC), "Project 704 – BWRVIP-06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals," dated May 24, 2002.

Enclosed are five (5) copies of the BWRVIP responses to the NRC Request for Additional Information on the BWRVIP document entitled "Section 4.0 of BWRVIP-06-A, BWRVIP Vessel and Internals Project: Safety Assessment of Reactor Internals" that was transmitted to the BWRVIP by the Reference 1 letter identified above.

Please note that the enclosed document contains proprietary information. Therefore, the request to withhold the BWRVIP-06-A report from public disclosure that was transmitted to the NRC by the Reference 2 letter identified above also applies to the enclosed document.

If you have any questions on this subject please call Bob Geier (Exelon, BWRVIP Assessment Committee Technical Chairman) by telephone at 630.657.3830 or by e-mail at robert.geier@exeloncorp.com.

Sincerely,

Rick Libra

Rick Libra
Exelon
Chairman, BWR Vessel and Internals Project

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BWRVIP RESPONSES to NRC REQUEST FOR ADDITIONAL INFORMATION on SECTION 4.0 of BWRVIP-06-A "BWRVIP VESSEL AND INTERNALS PROJECT: SAFETY ASSESSMENT of REACTOR INTERNALS"

Items from the Request for Additional Information on Section 4.0 of BWRVIP-06-A are repeated below followed by the BWRVIP response to the items.

RAI 06-A-1

In Section 4.1 of the BWRVIP-06-A report, the BWRVIP discusses the potential of loose parts resulting from the failure of steam dryers and separators that are carried with the steam. The discussion focused on migration of loose parts to prevent the main steam isolation valve (MSIV) and safety relief valve (SRV) performing their functions. Based on the recent industrial experience with the failure of steam dryers, the staff requests that the BWRVIP address whether large loose parts resulting from steam dryer failure can become a missile or cause damage to other reactor components (for instance, large loose parts entering reactor recirculation system (RRS) suction and causing damage of the recirculation pump).

BWRVIP Response to RAI 06-A-1

The discussion of steam dryer loose parts is not limited to the steam lines as suggested by RAI 06A-1, instead the reader is directed to the downcomer region discussion for other effects. [redacted]

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[redacted] Normal monitoring activities of the recirculation system would assure that either damage to the pump or resultant system flow reduction could be noted and appropriate steps taken to mitigate the condition.

RAI 06-A-2

In Section 4.1 of the BWRVIP-06-A report, the staff noted that the potential for impact on the standby liquid control system stand pipe is not included. The staff requests that the BWRVIP add this item to the list of concerns.

BWRVIP Response to RAI 06-A-2

Per GE Specification 21A8755 Rev. 4, the differential pressure and the standby liquid control line consists of two sections, one the standby liquid control line section serving as a pressure tap for one end of the core differential pressure instrumentation. The second pressure tap is an instrumentation line running to the top of the core plate. The differential pressure and the liquid control line are supported at the vessel nozzle, the shroud support skirt and along the shroud as it is routed inside the core. The resulting

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configuration is a pipe within a pipe that is well supported and is unlikely to be damaged from impact by a lost part entering the lower plenum. Even if the pipe attached to the core shroud was damaged by a loose part, it would not prevent the injection of sodium pentaborate into the vessel. This pipe is for core DP measurement and not standby liquid control. Additionally, failure of the pipe in the lower plenum would result in a change in measured core DP and be detected in the control room. See BWRVIP-06-A report Figures 2.2-1 through 2.2-4 that provides further detail of the various BWR 2-6 configurations.

RAI 06-A-3

In Section 4.1 of the BWRVIP-06-A report, the BWRVIP states "The geometry of certain parts and their components may be such that they would not be able to pass through the fuel bundle upper tie plate openings." The staff requests that the BWRVIP provide information regarding the size of the upper tie plate opening.

BWRVIP Response to RAI 06-A-3

The following table presents upper tie plate pass through sizes in millimeters for potential loose parts with rectangular, cylindrical and square shapes for various GE Nuclear Fuel (GNF) BWR fuel designs. BWRVIP will provide corresponding information for the other BWR fuel designs in a supplement to this RAI response.

Upper Tie Plate Pass through Sizes

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RAI 06-A-4

In Section 4.1.1 of the BWRVIP-06-A report, the BWRVIP discusses potential interference with MSIVs. The BWRVIP indicates that the fixed liner leg on the vertical centerline could be bent or broken when hit by a large loose part that might prevent the valve from closing. It appears that the other valve can also fail to close. The staff requests

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that the BWRVIP discuss the potential for both valves failing to close due to large loose parts being continuously generated as a result of the steam dryer failure.

BWRVIP Response to RAI 06-A-4

There is no known case where a redundant MSIV has failed to provide the design backup expected. BWRs are designed with redundancy as a standard for valves in a line that require the redundancy. It is highly unlikely that two large steam dryer loose parts would migrate down a single main steam line and bend or break both of the legs on the vertical centerline of both valves. Two valves were considered adequate for the original design, with approval being given based on the necessity of a two valve redundant design. One potential large lost part and definitely more than one lost part from a failed steam dryer should be detected by change in steam moisture quality, which is recommended to be monitored as discussed in BWRVIP-139. In addition, if a large loose part were to become lodged in a main steam line this would likely be detected as a mismatch in flow by control room indicators.

RAI 06-A-5

In Section 4.1.2 of the BWRVIP-06-A report, the BWRVIP indicates that potential loose parts are not expected to interfere with the SRV operation because of the following: (1) SRVs are closed during normal plant operation, and as such there is no flow to draw the loose parts into the stand pipe and the valve; and (2) the failure of system components coinciding with a loss of coolant accident (LOCA) is unlikely. Recent industrial experience has suggested that the shear layer, which separates the mean flow in the main pipe from the stagnant fluid in the branch, can be unstable due to the acoustic resonance in the SRV standpipe. Therefore, the staff requests that the BWRVIP address the possibility that the loose parts may be drawn into the standpipe and the valve due to vortex shedding and flow-excited acoustic resonances at the inlet.

BWRVIP Response to RAI 06-A-5

The SRVs for almost all plants are located on the top of the horizontal run of the main steam line piping with the standpipe oriented vertically. In this orientation, the inlet to the side branch located at the bottom of the standpipe and prevents the accumulation of condensate in the standpipe. This orientation will also ensure that loose parts cannot collect in the standpipe. The shear layer instability and vortex shedding associated with flow-excited acoustic resonances of the standpipe are local to the entrance region of the standpipe. The vortex acoustic coupling produces an in-out flow with rapid cycling (over 100 Hertz, typically, for a relief valve), and so the likelihood of drawing a loose part in without it subsequently being driven out into the Main Steam Line flow in the next half-cycle is very small. In order for the loose part to be drawn into the standpipe region, the loose part would have to be traveling along the top of the steam line past the standpipe opening, pass by the opening at the right time in the vortex cycle to be subjected to an upward drag force, and the vortex would have to be strong enough to lift the loose part up out of the main flow stream. Given the flow velocities in the main steam line, it is

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unlikely that anything other than the very smallest pieces could be lifted up out of the main stream and carried into the standpipe by the vortex. Once inside the standpipe, there is no force that will hold the loose piece in the standpipe where the fluid is stagnant; the cyclical nature of the vortex itself will tend to return the loose part to the main flow stream. Therefore, it is unlikely that the loose parts may be drawn into the standpipe by vortex shedding associated with flow-excited acoustic resonances in such a way that it can interfere with the SRV operation.

A few of the early plants may have safety or relief valves mounted on a vertical run of the main steam line with the standpipe oriented horizontally. In this configuration, it is possible that a small loose part may come to rest in the horizontal section of the standpipe. Again, in order for the shear layer instability and vortex shedding associated with flow-excited acoustic resonances to affect the trajectory of the loose part, the loose part would have to be traveling along the side of the steam line right next to the standpipe opening, at just the right time in the vortex cycle, and the part would have to be very small in order for the vortex to be able to deflect the part out of the main flow stream. It is unlikely that these conditions will be satisfied and, therefore, it is unlikely that the vortex shedding associated with flow-excited acoustic resonances will draw a loose part into the standpipe where it can interfere with the SRV operation.

RAI 06-A-6

In Section 4.1.2 of the BWRVIP-06-A report, the BWRVIP states, "Impact of a loose part along with loss of coolant accident [LOCA] or anticipated operational occurrence [AOO] is not considered in the loose parts analysis process. This is due to the fact that the failure of a system or component due to a loose part, coincident with LOCA or AOO is highly unlikely." Even though a LOCA is a low probability event, an AOO is an anticipated operational occurrence and hence, the staff believes that safety impact needs to be considered for the failure of a system or component due to a loose part. The staff requests that the BWRVIP address this issue.

BWRVIP Response to RAI 06-A-6

As discussed in RAI response 06-A-5, loose parts are not expected to come to rest within an SRV standpipe and therefore it would be incredible to assume the momentary occurrence of a passing loose part coincident with an AOO. The implication is that the AOOs (transients) need to be looked at assuming some flow blockage from a loose parts analysis. What is not done is to combine the effect of the flow blockage in meeting the consequences of the Accident or AOO, which is considered to be an incredible condition (blockage of the limiting fuel element to the maximum degree and the design basis event conditions). This is a significant difference from saying the effect of the loose part is not considered in the Accident and AOOs.

RAI 06-A-7

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The staff recommends that the BWRVIP address in Section 4.1.3.of the BWRVIP-06-A report the necessary measures that would provide early warnings prior to any occurrence of flow blockage of individual fuel channels due to loose parts. These early warnings would enable the licensees to take proper steps so that local overheating and, consequently, fuel damage to the affected fuel channels can be prevented.

BWRVIP Response to RAI 06-A-7

The GE Flow Blockage LTR (NEDO-10174) provides the basis for addressing fuel damage from flow blockage. See Reference 13 in Section 5 of BWRVIP 06 report. It is not possible to reliably detect flow blockage with standard BWR instrumentation before boiling transition occurs.

RAI 06-A-8

Regarding Section 4.1.4 of the BWRVIP-06-A report, the staff requests that the BWRVIP provide information regarding the mesh size for the debris filter on the lower tie plate.

BWRVIP Response to RAI 06-A-8

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RAI 06-A-9

In Section 4.1.5 of the BWRVIP-06-A report, the BWRVIP discusses the potential for impact damage on reactor vessel internals (RVI) components which focus on the loose parts transporting through the jet pump nozzles to lower plenum area. This section concludes that no significant damage to the RVI components is expected due to the very low core flow in the reactor lower plenum. The staff requests that the BWRVIP discuss the potential damage to the RVI components or instrumentation lines due to loose parts that are carried by the Feedwater flow to the reactor and down into the annulus area. This observation is based on the recent industrial experience in Dresden uprated power operation where feedwater probes were found penetrating the feedwater sparger inside the reactor.

BWRVIP Response to RAI 06-A-9

A potential lost part discharged from a feedwater sparger could migrate in the reactor annulus, where it might hit a jet pump sensing line or core spray header supply line, for which no damage is expected. The impingement by the part on the separator, shroud, shroud hardware, recirculation lines, jet pump assemblies, core spray piping, and other reactor internals would normally be of minimal to no consequence because of the relative small size of the part. In rare cases, a part may become stuck or trapped in an area like a

BWRVIP Proprietary Information

jet pump nozzle or in the jet pump assembly itself and cause wear of the equipment, hindering its expected performance. Performance deterioration would not be a design issue and would be something that can be monitored so that corrective action could be taken, if necessary. There are no thermal consequences with the water jet from a jet pump nozzle with fretted hole during normal operation because the water is the same temperature as the downcomer.

RAI 06-A-10

Based on the discussion provided in Section 4.1.6 of the BWRVIP-06-A report, the staff requests that the BWRVIP address possible crevice corrosion due to the presence of a crevice created between a lodged loose part and the RVI component. Crevice corrosion can be more pronounced in RVI components where effective protection due to hydrogen water chemistry and or noble metal chemical addition is not available and the reactor water flow is stagnant.

BWRVIP Response to RAI 06-A-10

As stated in BWRVIP-06-A, all loose parts are made of materials that have been approved for reactor use and have an established history of use in operating BWRs. These materials that are used in reactor internals are essentially austenitic stainless steel or austenitic nickel based materials. These materials form corrosion resistant oxides, which will slow down and minimize corrosion processes. Additionally, the low alloy steel pressure vessel is clad over much of the interior with austenitic corrosion resistant cladding.

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RAI 06-A-1 1

In Section 4.1.7 of the BWRVIP-06-A report, the BWRVIP discusses the potential for interference with the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) operation, indicating that during normal operation, both systems are idle and stagnant, and no flow will draw loose parts into the system piping. As requested previously, the staff requests that the BWRVIP address whether loose parts could be drawn into the system piping when the shear layer at the opening to the HPCI and RCIC systems become unstable due to a higher flow of the main steam line because of acoustic resonance.

BWRVIP Response to RAI 06-A-11

As discussed in response to RAI 06-A-5, in order for the shear layer instability and vortex shedding associated with flow-excited acoustic resonances to affect the trajectory of the loose part, the loose part would have to be traveling along the side of the steam line right next to the HPCI or RCIC opening, at just the right time in the vortex cycle, and the part would have to be very small in order for the vortex to be able to deflect the part out of the main flow stream. It is unlikely that these conditions will be satisfied and, therefore, it is unlikely that the vortex shedding associated with flow-excited acoustic resonances will draw a loose part into the HPCI or RCIC where it can interfere with the HPCI or RCIC operation.

RAI 06-A- 12

The BWRVIP in Section 4.1.1 1 of the BWRVIP-06-A report specifies Title 10 of the *Code of Federal Regulations* (1 0 CFR) Part 100, "Reactor Site Criteria," limits for off-gas radiation release. However, the staff notes that the main concerns are related to ALARA and hence 10 CFR Part 20, "Standards for Protection Against Radiation," is applicable. The staff requests that the BWRVIP address the application of 10 CFR Part 20 with regards to ALARA considerations. The staff also requests that the BWRVIP submit GE Service Information Letter (SIL) Number 552, "Fuel Failure Caused By Metal Debris."

BWRVIP Response to RAI 06-A-12

See attached SIL #552 Rev 1.

The subject statement concerns the ability of the offgas system to monitor and maintain offsite releases below plant specific licensing limits to comply with 10CFR100 requirements. One of the evaluation basis statements of the BWRVIP06-A report in Section 1.3 is "The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10CFR100 guidelines."

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All facilities must follow 10 CFR 20, "Standards for Protection Against Radiation," and are required to have an approved As Low As Reasonably Achievable (ALARA) program for work at the facility, it is expected that all work at the facility follows a procedure requiring that a necessary radiation survey is to be completed prior to the start of any work in a radiation environment. With a completed radiation survey, the worker is expected to keep the work radiation exposure absorbed ALARA. In an ALARA program there are four major ways to reduce radiation exposure from any radioactive lost part to workers or to the population:

- Shielding. Use proper barriers to block or reduce ionizing radiation.
- Time. Spend less time in radiation fields.
- Distance. Increase distance between radioactive sources and workers or population.
- Amount. Reduce the quantity of radioactive material for a practice

RAI 06-A-13

The staff notes that some licensees keep a list of the lost parts from the beginning of operation of the plant. The staff recommends that the BWRVIP add a section to the report to indicate that keeping an inventory and history of the lost parts (indicating the number, size, date the part was lost and the date it was recovered) is good practice to keep a track of the lost parts.

BWRVIP Response to RAI 06-A-13

Section 4.0 of BWRVIP-06A will be revised to indicate that keeping an inventory and history of the lost parts (indicating the number, size, date the part was lost and the date it was recovered) is good practice.

RAI 06-A-14

The staff requests that the BWRVIP reference the following document in Section 4.0 of the BWRVIP-06-A report: NRC Regulatory Guide 1.133, Revision 1, "Loose Parts Detection Program for the Primary Systems of Light Water Cooled Reactors," 1981.

BWRVIP Response to RAI 06-A-14

The BWRVIP believes that referencing NRC Regulatory Guide 1.133, Revision 1, "Loose Parts Detection Program for the Primary Systems of Light Water Cooled Reactors," 1981 in Section 4.0 of the BWRVIP-06-A report would lead to confusion in that a review of the reported operating history for the Lost Parts Monitoring System (LPMS) does not indicate significant differences in the impact or consequence of loose parts in the reactor coolant pressure boundary between plants with a LPMS and those without. In a letter addressed to Mr. James M. Kenny, Chairman BWR Owners' Group dated January 25, 2001; the NRC approved the Loose Parts Monitoring System regulatory

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relaxations that were requested by the BWROG in General Electric (GE) Topical Report NEDC-32975P, "Regulatory Relaxation for BWR Loose Parts Monitoring Systems" dated July 31, 2000. The January 25, 2001 approval letter and the associated design evaluation that defines the basis for NRC acceptance of the topical report is enclosed in that report.

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BWRVIP SUPPLEMENTAL RESPONSES TO NRC RAIS ON SECTION 4.0 OF BWRVIP-06-A

ELECTRIC POWER RESEARCH INSTITUTE

2008-169 BWR Vessel & Internals Project (BWRVIP)

June 12, 2008

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: Vanice Perin

Subject: Project No. 704 – BWRVIP Supplemental Responses to NRC Request for Additional Information on Section 4 of BWRVIP-06-A

Reference:
1. BWRVIP letter 2007-358 from Rick Libra (BWRVIP Chairman) to Document Control Desk (NRC) "Project No. 704 – BWRVIP Responses to NRC Request for Additional Information on Section 4 of BWRVIP-06-A," dated November 30, 2007.
2. BWRVIP letter 2002-138 from Carl Terry (BWRVIP Chairman) to Document Control Desk (NRC) "Project 704 – BWRVIP -06-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals," dated May 24, 2002.

Enclosed are five (5) copies of a supplement to the BWRVIP responses to the NRC Request for Additional Information submitted by the Reference 1. This letter provides supplemental information to BWRVIP responses to NRC RAIs numbered RAI 06-A-3 and RAI 06-A-8. Specifically, supplemental information is provided on upper tie plate pass through sizes and debris filter opening sizes for AREVA and Westinghouse fuel in BWR units.

Please note that the enclosed document contains proprietary information. Therefore, the request to withhold the BWRVIP-06-A report from public disclosure that was transmitted to the NRC by the Reference 2 also applies to the enclosed information.

If you have any questions on this subject please call Bob Geier (Exelon, BWRVIP Assessment Committee Technical Chairman) by telephone at 630.657.3830 or by e-mail at robert.geier@exeloncorp.com.

Sincerely,

Rick Libra

Richard Libra, Exelon
Chairman, BWR Vessel and Internals Project

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BWRVIP SUPPLEMENTAL RESPONSES to NRC REQUEST FOR ADDITIONAL INFORMATION on SECTION 4.0 of BWRVIP 06A "BWRVIP VESSEL AND INTERNALS PROJECT: SAFETY ASSESSMENT of REACTOR INTERNALS"

RAI 06-A-3

In Section 4.1 of the BWRVIP-06-A report, the BWRVIP states "The geometry of certain parts and their components may be such that they would not be able to pass through the fuel bundle upper tie plate openings." The staff requests that the BWRVIP provide information regarding the size of the upper tie plate opening.

BWRVIP Supplemental Response to RAI 06-A-3

Upper tie plate pass through sizes for potential loose parts with rectangular, cylindrical and square shapes were provided in the original response to RAI 06-A-3 for various GE Nuclear Fuel (GNF) BWR fuel designs. Additional upper tie plate pass through size information has been obtained for the other fuel designs (Westinghouse and AREVA) currently in use in BWRs as shown below.

Upper tie plate pass through sizes for AREVA and Westinghouse fuel used in BWRs are given below:

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RAI 06-A-8

Regarding Section 4.1.4 of the BWRVIP-06-A report, the staff requests that the BWRVIP provide information regarding the mesh size for the debris filter on the lower tie plate.

BWRVIP Supplemental Response to RAI 06A-8

The debris filter mesh opening size for GE fuel was provided in the original response to RAI 06-A-8. Additional debris filter mesh size information has been obtained for the

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other fuel designs (Westinghouse and AREVA) currently in use in BWRs as shown below.

Westinghouse Fuel Debris Filter

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AREVA Fuel Debris Filters

AREVA fuel incorporates two debris filter designs. Pass through sizes for each design are shown in the table below:

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RECORD OF REVISIONS

BWRVIP-06	Original Issue EPRI report TR-105707
BWRVIP-06-A	Information from the following documents was used in preparing the changes included in this revision of the report: "BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals (BWRVIP-06)", EPRI Report TR-105707, October, 1995 Letter from G.C. Lainas (NRC) to Carl Terry (BWRVIP) September 15, 1998: "Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-06 Report (TAC NO. M93926)" (98-397A) No changes to the report were required.
BWRVIP-06 Revision 1-A	Information from the following documents was used in preparing the changes included in this revision of the report: 1. BWRVIP letter from William Eaton (BWRVIP Chairman) to Document Control Desk (NRC), Attention; Meena Khanna, "Project No. 704 – Revised Section 4.0 "Consideration of Loose Parts" of BWRVIP-06-A: BWR Vessel and Internal Project, Safety Assessment of BWR Reactor Internals", May 11, 2005. (BWRVIP Correspondence File Number 2005-205). 2. Letter from Matthew A. Mitchell (NRC) to Bill Eaton (BWRVIP Chairman), "Request for Additional Information – BWRVIP-06-A: BWR Vessel and Internals Project Safety Assessment of BWR Reactor Internals", December 21, 2006. (BWRVIP Correspondence File Number 2006-516A) 3. BWRVIP letter from Rick Libra (BWRVIP Chairman) to Document Control Desk (NRC), Attention: Holly Cruz, "Project 704 – BWRVIP Responses to NRC Request for Additional Information on Section 4.0 of BWRVIP-06-A", November 30, 2007. (BWRVIP Correspondence File Number 2007-358). 4. BWRVIP letter from Rick Libra (BWRVIP Chairman) to Document Control Desk (NRC), Attention: Vanice Perin, "Project 704 –BWRVIP Supplemental Responses to NRC Request for Additional Information on Section 4.0 of BWRVIP-06-A", June 12, 2008. (BWRVIP Correspondence File Number 2008-169). 5. Letter from Mark J. Maxin (NRC) to Rick Libra (BWRVIP Chairman), Safety Evaluation for Electric Power Research Institute (EPRI) Boiling Water Vessel and Internals Project (BWRVIP) Topical Report (TR)-1006598, "BWR Vessel and Internals Project – BWRVIP-06-A, Safety Assessment of BWR Reactor Internals, Revised Section 4.0: Consideration of Loose Parts" (TAC NO. MC7448), July 29, 2008. (BWRVIP Correspondence File Number 2008-205A) Details of the revisions can be found in Table E-1.

Table E-1
Revision details

Required Revision	Source of Requirement for Revision	Description of Revision Implementation
Add NRC Safety Evaluation (SE) behind report title page.	NRC Request.	Added NRC SE on BWRVIP-06 Revision1.
Add BWRVIP (NEI-03) Implementation Requirements.	BWRVIP-94 Revision 1 requirement.	Added new Section 1.4 "Implementation Requirements".
Needed to update Section 4 to expand discussion of effect of loose parts on several systems	Concern over effect of loose parts stemming from steam dryer damage experienced by the Quad Cities units.	Replaced Section 4 "Consideration of Loose Parts" in its entirety.
Add a note that it is good practice to keep and inventory and history of lost parts.	BWRVIP response to NRC RAI 06-A-13.	Added a sentence in the first paragraph of Section 4.0 noting that it is good practice to keep an inventory and history of lost parts.
Add limitation that the revised Section 4.0 is applicable only to BWR/2-6 plants.	NRC SE Section 4.0.	Added new Section 4.3 listing limitation that the revised Section 4.0 is applicable only to BWR/2-6 plants.
Add requirement that plant specific assessment is required if a loose part is detected.	NRC SE Section 4.0.	Added new Section 4.3 listing requirement that plant specific assessment is required if a loose part is detected.
Added remainder of NRC Correspondence in Appendices B-D.	NRC Request.	Added new Appendices B-D.
Consolidate information in previous Appendix B into a new Appendix E "Record of Revisions".	Editorial.	Deleted old Appendix B and added new comprehensive Appendix E "Record of Revisions".
End.		

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