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A PROGRAM FOR RISK ASSESSMENT ASSOCIATED WITH IGSCC OF BWR VESSEL INTERNALS[1]

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ABSTRACT

A program is being carried out for the United States Nuclear Regulatory Commission (NRC) by the Idaho National Engineering and Environmental Laboratory INEEL), to conduct an independent risk assessment of the consequences of failures initiated by intergranular stress corrosion cracking (IGSCC) of the reactor vessel internals of boiling water reactor (BWR) plants. The overall project objective is to assess the potential consequences and risks associated with the failure of IGSCC-susceptible BWR vessel internals, both singly and in combination with the failures of others, with specific consideration given to potential cascading and common mode effects on system performance. This paper presents a description of the overall program, including a completed preliminary qualitative risk assessment, and a program that is underway to modify an existing probabilistic risk assessment (PRA) of a BWR/4 plant to include IGSCC-initiated failures, subsequently to complete a quantitative PRA.

INTRODUCTION

General Design Criteria 2 and 4 require that commercial nuclear reactor structures, systems, and components important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, and the effects of postulated accidents, including loss-of-coolant accidents (LOCAs). Boiling water reactor (BWR) internals components were originally believed to have been designed to accommodate these requirements. However, intergranular stress corrosion cracking (IGSCC) degradation has been observed in both core shrouds as well as a number of other BWR reactor internals components, many of which are important to plant safety. Fig. 1 is an isometric illustration of a typical BWR/3-4 design showing the reactor internals components.

Although IGSCC of reactor internals had been recognized for over 20 years, this phenomenon received increased attention, beginning when crack indications were reported at core shroud welds located in the beltline region of an overseas BWR in 1990. The core shroud is a stainless steel cylinder that is located inside the reactor vessel. It serves to both provide lateral support to the reactor core and to direct the flow of water inside the reactor vessel, and is generally regarded as component whose integrity is critical to maintaining core safety. Later, a visual inspection of a U.S. BWR core shroud revealed crack indications at several weld regions. Subsequently, General Electric (1993a, 1994a) and the NRC (1993a, 1994a, 1994b) issued correspondence regarding core shroud cracking.

In addition to the BWR core shroud degradation, other BWR reactor internals components, including shroud support access hole cover welds, jet pump hold-down beams, core spray systems, and top guides have also been experiencing IGSCC degradation over the years (Medoff, 1996). These instances have for the most part been sporadic, were not believed to be of major safety importance, and were addressed by General Electric (1978, 1979, 1980, 1981, 1986a, 1986b, 1988, 1993b, 1994a, 1995) through notices such as Safety Information Letters (SILs) and by the NRC through Information Notices and a Bulletin that have been issued from time-to-time since about 1980 (NRC, 1980a, 1980b, 1988, 1992, 1993b, 1995, 1997). However, the instances of core shroud cracking served to escalate attention as to the seriousness of the IGSCC problem in BWR reactor internals.

The NRC Office of Nuclear Reactor Regulation has followed the problem and issued a Bulletin, Information Notices, and Safety Evaluation Reports (SERs), as well as a Generic Letter (NRC, 1994b). The primary emphasis has been placed on core shroud degradation, but common mode or cascading failure of other components could also have safety significance. Consequently, this NRC-sponsored program described in this paper has been initiated to conduct a risk

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assessment investigating the concern of cascading failures of BWR vessel internals.

The objective of the study is to assess the potential consequences associated with the failure of IGSCC-susceptible BWR reactor internals components, both singly and in combination with the failures of others. Specific consideration is given to potential cascading and common mode effects on system performance stemming from cracking of core shrouds and other BWR reactor internals components when subjected to design-basis and beyond-design-basis accident-loading conditions such as seismic events.

The focus is on mechanical design, failure locations, consequences, potential accident scenarios, and characterization of risk associated with IGSCC degradation of BWR vessel internals. The scope is limited to the basic risk evaluation, including the following:

- The only degradation mechanism considered in this study is IGSCC, including contributing SCC mechanisms such as irradiation-assisted SCC (IASCC). It is recognized that other degradation mechanisms such as fatigue can act synergistically with IGSCC in that a crack which is initiated by IGSCC can propagate to failure from fatigue.
- The NRC is investigating the causes and contributing aspects of the IGSCC problem in separate programs. Industry groups are investigating inspection, mitigation, repair, or replacement, as well as the causes and contributing aspects of IGSCC.
- There are five currently operating types of BWRs in the U.S., designated BWR/2 through BWR/6. Since there is only a single BWR/1 in operation, Big Rock Point, and it is expected to be decommissioned in the near future, the scope of work in this study does not include the unique BWR/1.

BACKGROUND

A brief summary of background information on the IGSCC problem in BWR reactor internals is presented in this section.

Causes of IGSCC

The three basic elements that must *all* be present for IGSCC to occur are:

1. a susceptible material
2. a chemically aggressive environment
3. a high tensile stress

Under normal circumstances, the stress must be above the yield stress, which can occur at locations such as residual stresses around welds. However, if certain other factors are present, the conditions for the three basic elements listed above (such as the need for the tensile stress to be above the yield stress), may be somewhat altered.

Material Types 304 and 316 stainless steels, when the material composition in the microstructure between the interiors of the individual grains and the grain boundaries is nonuniform, are susceptible materials. Type 304 stainless steels contain alloying elements of 18% chromium and 8% nickel, and Type 316 stainless steels contain 16% chromium, 12% nickel, and 2% molybdenum. The chromium would normally make the steels highly resistant to SCC. However, certain heat treatments and welding processes can cause chromium depletion at the grain boundaries, and the intergranular area becomes susceptible to IGSCC. High nickel-based steels such as Alloys 600 and X-750 are also susceptible.

Environment During normal operation, the BWR coolant is at a temperature of about 288°C (550°F), and contains a few hundred ppb dissolved oxygen and other oxidizing products resulting from the radiolytic decomposition of the coolant, as well as various levels of ionic impurities. The products of the radiolytic decomposition include oxygen, hydrogen peroxide, and small amounts of water, hydrogen, and hydroxide ions. The relative concentrations of these products determine the degree of aggressiveness of the coolant, and thereby the rate of IGSCC formation. Although the normal oxygen content in the BWR coolant is a few hundred ppb, the oxygen concentration may approach thousands of ppb during startup if air ingress takes place.

Stress The third contributing factor is a high tensile stress. With no contributing factors, the localized stress must be above the yield stress to initiate IGSCC. In general, the higher the stress, the less time it takes for IGSCC to initiate. Welding and fitup stresses can be in excess of the material yield stress. Operating stresses and bolt preloads also contribute to the stress state. Locations of stress concentration, such as threads or holes, can intensify the stresses.

Factors that Enhance IGSCC

In addition to the three major causes of IGSCC, there are other related factors that can promote IGSCC, such as (Brown and Gordon, 1987):

- high electrochemical potential (ECP) and coolant conductivity (above a threshold of about 0.3 μS/cm)
- the presence of crevices
- IASCC (above a cumulative neutron threshold of about 5×10^{20})
- cold work (above a threshold of about 20%)

IGSCC Aging Management Methods

Aging management methods for IGSCC in BWRs can be categorically grouped as:

- inspection
- monitoring
- mitigation
- modification or repair
- replacement

The first two methods are used to assess the IGSCC problem. Inspection plans involve both visual and volumetric techniques. GE has developed individual methods for certain susceptible locations; for example, specialized equipment has been developed for inspection of the top guides, access hole covers, and incore monitor housings. Eddy-current testing has been used to inspect the incore monitor housing-to-vessel lower head welds. These inspections alert the licensee as to the presence and extent of IGSCC, but do nothing to correct the problem. They provide the basic information needed to know when to repair or replace the degraded component, and give researchers field knowledge of IGSCC so that they can develop laboratory tests and propose initiation models for understanding the phenomenon.

The basic ASME Code inspection guidelines for reactor internals are in Table IWB-2500-1 of Section XI. However, since the basic Code requirements are inadequate for IGSCC detection in reactor internals, more extensive examination guidelines for each reactor internals component have been published by EPRI (1995).

The final three aging management methods are solutions to the problem. Mitigation techniques alter the three basic causes of IGSCC (the material, the environment, or the stress), or they may change the design so that crevices are removed from the susceptible location. The susceptibility of the material may be reduced by using a more corrosion-resistant composition such as 316L stainless steel, or by changing the heat treatment of the material. The material may be protected by coating the component with a corrosion-resistant substance. Noble metals are a candidate for such coatings, applied either directly, or by injecting them into the coolant and allowing them to plate out on the exposed surfaces of the reactor internals.

The potential for corrosion of the environment may be reduced by control of water chemistry to remove ionic impurities, and by the use of hydrogen water chemistry (HWC). HWC involves injecting hydrogen into the feedwater to reduce the oxidizing products in the coolant (primarily O_2 and H_2O_2), and thus reduce the ECP of the stainless steel components. The hydrogen is injected through taps in the suction line leading to the condensate booster pump or to the main feedwater pumps. The amount of hydrogen injected depends on the plant design. Oxygen levels as low as 5 ppb are reported to have been achieved using HWC. Many plants are using HWC, and it is GE's goal that all BWR plants switch to HWC (Jones and Nelson, 1990).

Stress levels have been reduced by making the cross-section of a component larger or by reducing preload forces, as has been achieved the in case of jet pump holddown beams. Components have been redesigned to eliminate crevices, as in the case of BWR/6 top guides.

A degraded component may be modified, repaired or replaced, either when it has degraded to the point when the end of its safe, useful life has been reached, or as a preventative measure. GE has or is developing repair and/or replacement methods for many of the reactor internals. For example, for the core shroud, a modification consists of shroud clamping devices (designs have been developed by two different vendors).

INITIAL PROGRAM PHASE

The first phase of the study involved acquiring and evaluating relevant background information on IGSCC of BWR reactor internals, to qualitatively assess potential accident scenarios, and to identify scenarios for detailed analyses, that is, those expected to have large effects on Core Damage Frequency (CDF). This phase has been completed.

Accident Sequences

Differences in reactor internals designs and accident mitigation systems for the various BWR types were categorized, the degradation of BWR internals to date was catalogued, and aging management practices were reviewed. From this background study, various types of *systems* failure modes that could result from simultaneous common mode failures of various combinations of reactor internals were catalogued. This included the consideration of functional losses or significant degradations of certain inside-reactor vessel systems. A similar assessment for various types of *mechanistic* failure modes (i.e., the potential results of physical impacts/interactions, common mode and cascading failures, etc., between various reactor internals components due to failures and/or degradations of the components that are subject to IGSCC degradation), was made. This included the various ways that these components might fail and how those types of failures might affect other components inside the reactor vessel. Approximately 250 different and unique scenarios were identified.

The safety significance was also evaluated. This is generally component specific; however, one common safety significance is a loose part which can result from IGSCC. There are three basic safety consequences:

1. a loose part can inhibit control rod motion
2. a loose part can block or partially block a coolant flow channel
3. a large loose part can impact adjacent components and impair their function

There are other safety consequences not associated with loose parts which could be caused by damage to any of several reactor internals components, such as:

1. increased coolant leakage between plenums
2. damage to emergency coolant or shutdown systems [for example, the standby liquid control (SLC) system]
3. damage to control rods or prevent their motion
4. cause of a reactor coolant system leak

In order to reduce the number of scenarios to be considered, a screening logic based on the safety consequences identified above was prepared (Fig. 2). Five criteria were developed that are believed to cover the most important issues necessary to adequately address public safety with regards to reactor vessel internals failures. The screening logic was applied to all 250 initially identified scenarios. Of these, 148 remained after the screening, which reduced the work scope somewhat, but still left a large number of sequences to evaluate. A quantitative risk assessment was subsequently conducted on these remaining 148 sequences, as described in the following section.

Preliminary Qualitative Risk Assessment

A qualitative ranking (based on potential contributions to CDF) was made of potential accident scenarios which can be exacerbated by IGSCC degradation of reactor internals. Various possibilities of single, common mode, and cascading failure sequences were postulated for the high, medium, or low rankings. Although the rankings were qualitative, a NUREG-1150 PRA was used to assist in providing for an estimate for the rankings.

A preliminary risk assessment including a list of potential safety concerns (i.e., possible accident scenarios), deterministically developed, was made. Specific areas were described for each potential accident sequence where additional analyses are needed to provide a more definitive understanding of accident scenarios that involve either simultaneous (i.e., common mode) or cascading (i.e., sequentially caused by other failures). The scope included both deterministic failure considerations and qualitative risk assessments.

Most (about 100) of the scenarios were ranked high. Although there appears to be a large number of scenarios, there is a great deal of redundancy in that the high-ranking scenarios fall into variations of two basic categories:

1. loss of the reactor protection system (RPS)
2. loss of coolant to the core

The high-ranking categories were broken into subcategories to further differentiate between the various causes of loss of the RPS and the various ways that coolant to the core could be lost. The following seven subcategories were chosen.

1. loss of scram capability
2. standby liquid control (SLC) system nonfunctional
3. both RPS and SLC nonfunctional
4. medium LOCA with loss of SLC
5. both high-pressure coolant injection (HPCI) and low-pressure coolant injection LPCI ineffective [no redundant emergency core cooling system (ECCS)]
6. core reflood to two-thirds level cannot be maintained (treated as loss of ECCS)
7. high-pressure coolant system (HPCS) and SLC (through sparger) eliminated (several BWR/5 and BWR/6 plants)

Each of the 100 high-ranked scenarios can be placed into one or more of these subcategories.

CONTINUED PROGRAM PHASE

The initial phase of the project, which was primarily to scope the overall effort, has been completed. The second phase is now underway, in which quantitative calculations will be performed. However, there are many difficulties in carrying out this program, such as:

(a) the large number of components and failure sequences
(b) the different types of BWRs
(c) the difficulty in estimating crack sizes and growth rates
(d) there are a large number of disciplines involved
(e) there are limited "good" PRA and thermal-hydraulic models available

It was decided to narrow the research scope to provide a simplified, cost-effective approach. The following simplifications were proposed as an initial approach:

(a) select a single plant for study
(b) select a single component and probable failure locations for initial calculations
(c) perform minimal calculations and research
(d) develop a methodology to introduce IGSCC-induced failures into an existing PRA which can then be applied to the failure of any BWR vessel internals component
(e) convert an existing TRAC-B model to a representative plant to determine flow characteristics
(f) estimate the failure probabilities and insert events associated with the failure of the selected component into an existing PRA, considering a single failure at the most likely locations, common mode failures, and cascading failure sequences. If successful, then apply to other components.
(g) use expert panels to critique methods, offer suggestions for approach, and help in estimating probabilities and uncertainties

Plant Selection

A number of criteria were used to select a plant for study, including:

(a) is there an existing PRA model for the plant (internal and external events)
(b) is there an existing TRAC-B model for the plant (or one for a similar plant)
(c) is the plant typical
(d) older plants were preferred

A BWR/4 was chosen for study. As a high-power BWR/4, it is the most representative type of BWR. There is a fairly good (but not ideal) PRA, and there exists a BWR/4 TRAC-B thermal-hydraulics model which can be modified to represent the selected plant.

Initial Component Selection

A number of criteria were used to select a specific reactor internals component for study, including:

(a) degradation to date
(b) cascading possibilities
(c) safety significance
(d) typicality

The jet pump was the reactor internals component selected for study, as there has been recently discovered cracking in jet pump riser inlet welds (NRC, 1997) and jet pump failure could lead to a variety of cascading failure sequences. IGSCC failures have also initiated at the jet pump holddown beams, the first instance occurring in a BWR/3 in 1980. Cracking also was found in the beams of two BWR/6 plants. Subsequently, the beams have been redesigned.

PRA Modification

The existing PRA for the selected plant is being modified for the IGSCC-induced failures. The following are examples of modifications being made:

1. introducing information from the latest Individual Plant Evaluation (IPE)
2. modifying component seismic fragilities to reflect more recent seismic hazard fragility information
3. including internal and external event PRA branches that were not used in previous PRA studies because of low probabilities
4. separating the main steam line, main feedwater line, and recirculation line breaks into three individual events, each with its own probability of occurrence
5. adding an event tree for IGSCC-induced initiating events

These modifications are applicable to all of the potential IGSCC-induced failures of reactor internals. However, the model will first be run only with the probabilities of IGSCC-induced failures of jet pump components included.

SUMMARY

A program is underway to perform an independent risk assessment of accident sequences initiated by IGSCC-induced failures of BWR reactor internals components. An initial phase has been completed in which background material was gathered and evaluated, potential accident sequences were identified, and a qualitative PRA was performed to rank the sequences as having a high, medium, or low potential to significantly change the core damage frequency. A second phase is underway to perform a simplified, quantitative PRA on a representative high-power BWR/4. The existing PRA for the plant is being upgraded and modified for the project, including introducing an event tree associated with reactor internals failures. Failures associated with jet pumps will be addressed first. If the methodology proves successful, the study will be extended to other major reactor internals components.

REFERENCES

Brown, K.S., and G.M. Gordon, 1987, "Effects of BWR Coolant Chemistry on the Propensity for IGSCC Initiation and Growth in Creviced Reactor Internals Components", *Third International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Traverse City, Michigan, August 30-September 3, 1987*, National Association of Corrosion Engineers, Houston, Texas, pp. 243-247.

Electric Power Research Institute, 1995, *BWR Vessel and Internals Project, Reactor Pressure Vessels and Internals Examination Guidelines (BWRVIP-03)*, EPRI TR-105696.

General Electric Company, 1978, "Removal of Startup Source Holders for the Reactor Pressure Vessel", SIL 215, Rev. 1.

General Electric Company, 1979, "Core Spray Piping Visual Inspection", SIL 289; Supplement 1, Rev 1, 1989.

General Electric Company, 1980, "Jet Pump Beam Cracks", SIL 330; Supplement 1, 1981.

General Electric Company, 1981, "Control Blade Lifetime", SIL 157, Rev. 2.

General Electric Company, 1986a, "Inspection of SRM/IRM Dry Tubes", SIL 409, Rev. 1.

General Electric Company, 1986b, "Shroud Head Bolt Cracks", SIL 433, February 7; Revision 1, Sept. 15, 1993.

General Electric Company, 1988, "Shroud Support Access Hole Cover Cracks", SIL 462; Supplement 1, 1989; Supplement 2, August 1990; Supplement 2, Rev. 1, December 31, 1990; Supplement 3, 1992.

General Electric Company, 1993a, "Core Shroud Cracks", SIL 572, Rev. 1, November 22.

General Electric Company, 1993b, "Jet Pump Beam Cracking", RICSIL 065, December 3.

General Electric Company, 1994a, "Top Guide and Core Plate Cracking", RICSIL 071, November 22.

General Electric Company, 1994b, "BWR Core Shroud Evaluation", GENE-NE-523-148-1193, April.

General Electric Company, 1995, "Top Guide and Core Plate Cracking", SIL 588, February 17.

Jones, R.L., and J.L. Nelson, 1990, "Hydrogen Water Chemistry for BWRs: A Status Report on the EPRI Development Program", *Nuclear Engineering and Design 124*, North-Holland, Amsterdam, pp. 33-42.

Medoff, J., 1996, *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, NUREG-1544.

United States Nuclear Regulatory Commission, 1980a, Information Notice 80-07, "BWR Jet Pump Assembly Failure", April 4.

United States Nuclear Regulatory Commission, 1980b, IE Bulletin 80-13, "Cracking in Core Spray Spargers", May 12.

United States Nuclear Regulatory Commission, 1988, Information Notice 88-03, "Cracks in Shroud Support Access Hole Cover Welds", February 2.

United States Nuclear Regulatory Commission, 1992, Information Notice 92-57, "Radial Cracking of Shroud Support Access Hole Cover Welds", August 11.

United States Nuclear Regulatory Commission, 1993a, Information Notice 93-79, "Core Shroud Cracking at Beltline Region Welds in Boiling Water Reactors", September 30.

United States Nuclear Regulatory Commission, 1993b, Information Notice 93-101, "Jet Pump Hold-Down Beam Failure", December 17.

United States Nuclear Regulatory Commission, 1994a, Information Notice 94-42, "Cracking in the Lower Region of the Core Shroud in Boiling Water Reactors", June 7; Supplement 1, July 19.

United States Nuclear Regulatory Commission, 1994b, Generic Letter 94-03, "Intergranular Stress Corrosion Cracking in Boiling Water Reactors", June 25.

United States Nuclear Regulatory Commission, 1995, Information Notice 95-17, "Reactor Vessel Top Guide and Core Plate Cracking", March 10.

United States Nuclear Regulatory Commission, 1997, Information Notice 97-02, "Cracks Found in Jet Pump Riser Assembly at Boiling Water Reactors", February 6.

NOTICE

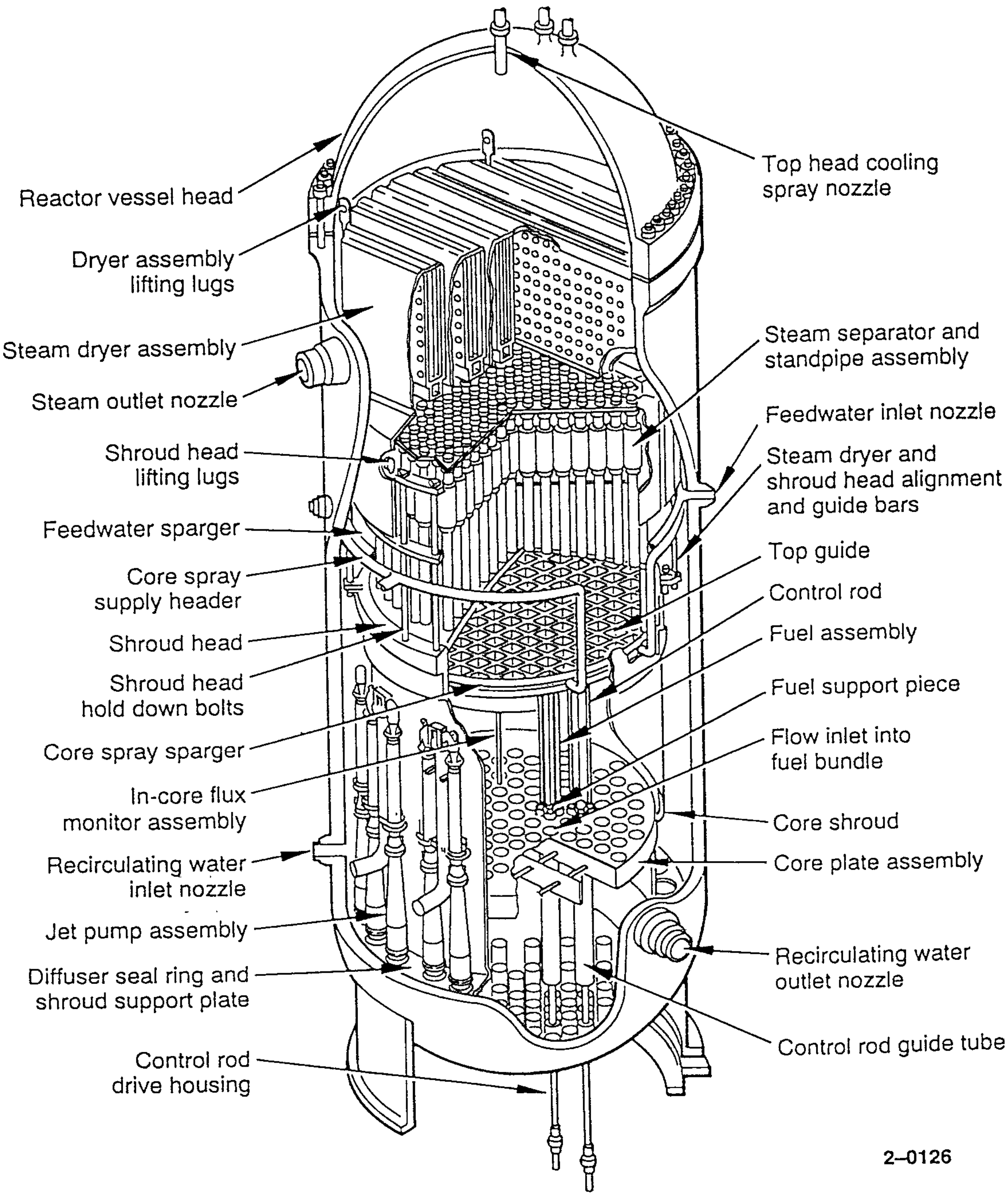

Fig. 1. BWR/3 or BWR/4 reactor vessel internals

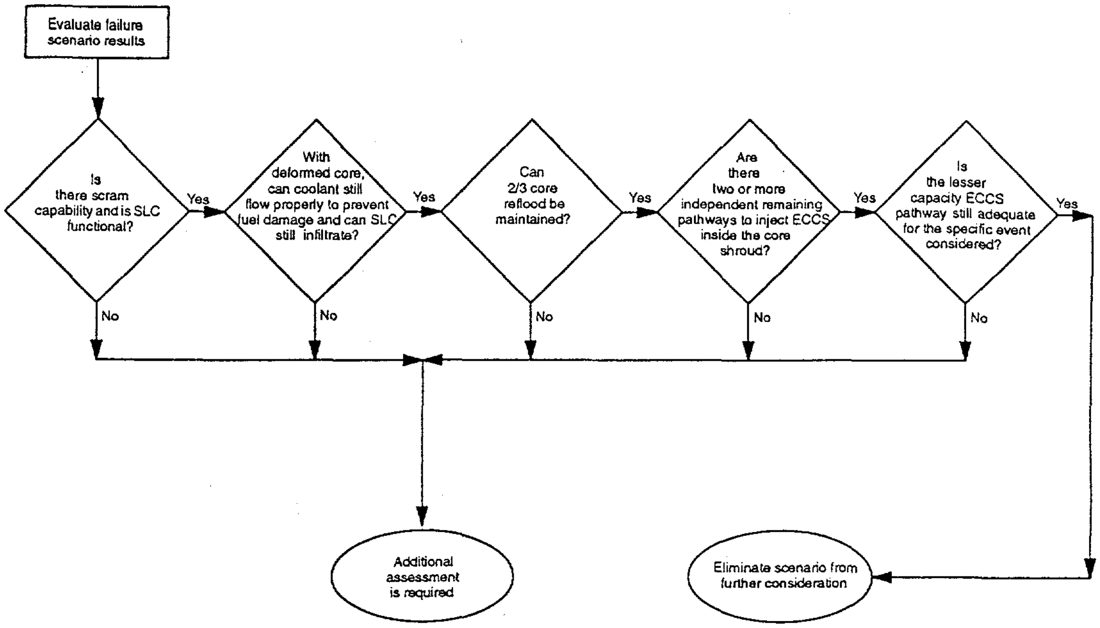

Fig. 2. Screening logic for the elimination of low-safety impact sequences during initiating events