SKI Report 2007:37

Research

Identification of Common Cause Initiating Events Using the NEA IRS Database

Rev 0

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SKI-perspective

Background

The study carried out within the framework of this project is a continuation of work conducted for SKI in 1998 on the identification of Common Cause Initiators based on operational events documented in the NEA Incident Reporting System (IRS). This project covered the events reported in the IRS database with the incident date in the period of 1980-01-01 – 2006-11-15.

The events of highest concern in nuclear power plants today are the dependent events, where a single event or a single cause initiate a disturbance with impact across redundant systems and, indeed, throughout a plant. Several such events have been observed in the past, often related with support systems, electrical systems, etc.

Dependent events are usually ranked the highest on the safety significance scale, due to their potential impact on the risk. Risk contribution from independent (random) events is typically less significant and generally easier to assess.

Among groups of dependent events that are occurring in NPP:s, a particular group of initiating events, called Common Cause Initiators, is of special interest. Those events are not just causing a disturbance in plant operation, but also degrade or even disable the function of a safety system that is needed to cope with the disturbances. Such events are often traced back to support systems, electrical distribution and I&C systems, secondary impact (pipe breaks), etc. Those are the areas where today's plants may still be vulnerable or have an unrevealed safety deficiency. Moreover, most of the today's plant specific PSA are relatively weak in modelling CCI:s, thus potentially neglecting an important risk contributor.

Scope

Based on the new operational experience accumulated in IRS in the period 1995-2006, the project focuses on the identification of new CCI events. An attempt is also made to compare the observations made in the earlier study with the results of the current work, in particular regarding the potential dependency mechanisms and plant systems/areas that are most susceptible to dependent failures.

International event reporting systems are important sources of information on problems related with CCIs. Events reported there are usually those that are judged to be the most serious ones, and may be containing the information on actual events or interesting precursors. By performing deeper and systematic evaluation of these occurred events, the knowledge and behaviour about these events are enhanced and for many persons new lessons to be learnt.

Result

Events of CCI type occur in operational NPP:s. Majority of operational events documented in the IRS database are complex events that include many occurrences both consequential and random. Most of these events are accident precursors. The events that involve consequential (dependent) degradation of the mitigation systems (the required attribute of CCI) are relatively rare events.

The analysis of operational events provided useful engineering insights regarding the potential dependencies that may originate in CCI:s. Some indications were also obtained on the plant Systems Structures Components/areas that are susceptible to common cause failures. Some of the following observations have been made:

- Direct interrelations between the accident mitigation systems through common support systems. This observation made in the earlier CCI study was confirmed in the current project
- The most important contributors of this type are electrical power supply systems and I&C systems. One of the mechanisms contributing to the degradation of the in-house electrical power supply systems is associated with malfunctions or spurious operation of the equipment protection devices/systems.
- Area-related events such as internal fires, flooding, water spray, and steam jet have also been found to be important sources of dependency.
- Another dependency mechanism is related to human factors, non-conservative planning of maintenance (one of the issues identified in the project) and errors of commission.

Impact on the operation of SKI

This report is an update of an earlier SKI Report 1998:9. Some earlier observations made, still confirm the presence of dominant dependency mechanisms involved in CCI events. Knowledge about repeating failures and failure mechanisms, there occurrence rates and trends. If these events are of the dependent type, they must be an important source of information in the regulatory and safety work. Lessons learnt will be widely spread to assert for an enhanced use of operating experience feedback in e.g., regulatory matters.

Continuing work within the research area

There are still needs for deeper analysis of some very tricky events in the AIRS database. Mostly due to the fact that this information source still summarize events on a rather general level. Detailed information about the events has to be found from experts at the origin of the plants.

Project information

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- SKI/RA-023/97
- SKI Report 1998:9, Identification of Common Cause Initiators in IRS Database, February 1998.

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Summary

The study presented in this report is a continuation of work conducted for SKI in 1998 on the identification of Common Cause Initiators (CCIs) based on operational events documented in the NEA Incident Reporting System (IRS). Based on the new operational experience accumulated in IRS in the period 1995-2006, the project focused on the identification of new CCI events. An attempt was also made to compare the observations made in the earlier study with the results of the current work. The earlier study and the current project cover the events reported in the IRS database with the incident date in the period from 01.01.1980 to 15.11.2006.

The review of the NEA IRS database conducted within this project generated a sample of events that provides insights regarding the Common Cause Initiators (CCIs). This list includes certain number of 'real' CCIs but also potential CCIs and other events that provide insights on potential dependency mechanisms.

Relevant characteristics of the events were analysed in the context of CCIs. This evaluation was intended to investigate the importance of the CCI issue and also to provide technical insights that could help in the modelling the CCIs in PSAs. The analysis of operational events provided useful engineering insights regarding the potential dependencies that may originate CCIs. Some indications were also obtained on the plant SSCs/areas that are susceptible to common cause failures.

Direct interrelations between the accident mitigation systems through common support systems, which can originate a CCI, represent a dominant dependency mechanism involved in the CCI events. The most important contributors of this type are electrical power supply systems and I&C systems.

Area-related events (fire, flood, water spray), external hazards (lightning, high wind or cold weather) and transients (water hammer, electrical transients both internal and external) have also been found to be important sources of dependency that may originate CCIs.

Abbreviations

(Applicable to the main report and appendices)

AC	Alternate Current
AFW	Auxiliary Feedwater
CCF	Common Cause Failure
CDF	Core Damage Frequency
CCI	Common Cause Initiator
CCS	Component Cooling System
CR	Control Rod
CS	Condensate System
CSI	Core Spray Injection
CCW	Condenser Cooling Water
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FMEA	Failure Mode and Effect Analysis
EFW	Emergency Feedwater
ELUPS	Emergency Lighting Uninterruptible Power System
EPS	Electrical Power Supply (house load)
ESD	Enhanced Shutdown System (GCR plant)
ESFAS	Engineered Safety Features Actuation System
HE	Heat Exchanger
HELB	High Energy Line Break
HP	High Pressure
HV	High Voltage
IE	Initiating Event
LOCA	Loss of Coolant Accident
LOOP	Loss Of Off-site Power
LP	Low Pressure
MCR	Main Control Room
MFW	Main Feedwater System
MS	Main Steam
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
P CCI	Potential CCI

PCS	Power Conversion System
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RCS	Reactor Coolant System
RHR	Residual Heat Removal
ROP	Regional Overpower (LHWR plant)
RPVHS	Reactor Pressure Vessel Head Spray (BWR)
SI	Safety Injection
SLB	Steam Line Break
SPC	Suppression Pool Cooling (BWR)
SSC	Structures, Systems and Components
SW(S)	Service Water (System)
VVER	PWR of Soviet design
ULSD	Ultra-low-sulphur diesel

Introduction

The study carried out within the framework of this Project is a continuation of work conducted for SKI in 1998 on the identification of Common Cause Initiators based on operational events documented in the NEA Incident Reporting System (IRS) [1, 2]. This work covered the events reported in the IRS database with the incident date in the period of 01.01.1980 – 31.12.1995. The current Project addresses the events that occurred between 01.01.1995 and 15.11.2006[1].

The objectives and scope of the both projects are similar. Also similar is the approach to IRS database search and methodology for the analysis of events. These subjects are briefly described in the following sections of this report.

Common Cause Initiators – Technical Background

The Common Cause Initiators are the operational events that cause a disturbance in plant operation requiring some form of controlling or mitigation action and also degrade or even disable the function of a safety system that is needed to cope with disturbances. Such events involve dependencies of various types and are often traced back to support systems, electrical distribution and I&C systems, secondary impact (pipe breaks), etc.

These events are of high safety concern in nuclear power plants. Dependent events are usually ranked the highest on the safety significance scale, due to their potential impact on the risk. These events are also more difficult to model in probabilistic safety assessment (PSA). Risk contribution from independent (random) events is typically less significant and generally easier to assess.

The project conducted for SKI in 1998 [1, 2] showed that such dependent events occurred in the plant operation history. The analysis of events collected in the IRS database provided useful insights regarding the dependency mechanisms and systems or plant areas susceptible to dependent failures. These insights help in understanding the subject of CCIs and, in consequence, in the modelling of such dependencies in PSA.

Considering the importance of CCI events the SKI decided to repeat a study similar to that performed in 1998. It focuses on the identification of actual operational occurrences of CCIs type on the basis of international operational experience collected in the IRS database. In particular, the project is intended to provide guidance on where more investigations may be warranted to enhance the considerations and the modelling of CCIs in PSA.

[1] There is an overlapping period of one year to ensure that no events are missed due to potential delay in the reporting process as well as changes in the IRS database organization and/or numbering

Project Objective

The objective of this Project is similar to the earlier project conducted for SKI in 1998 [1, 2]. It concentrates on gathering practical insights relevant for the identification of Common Cause Initiators (CCIs) based on event data available in the NEA Incident Reporting System (IRS) in the period of 1995 – 2006. The project is intended to investigate the applicable dependency mechanisms, type of direct and root causes of dependent occurrences, and systems/plant areas that are typically involved in CCI events. Insights from this investigation are expected to improve the understanding of CCIs, and, in consequence, their consideration in safety assessment of nuclear power plants and in particular plant specific PSA.

Scope

The scope of the study is the same as that of the earlier pilot study on CCI related issues [1, 2] which generated a list of CCI candidates and provided some practical insights that could help improving the understanding of the issue.

Based on the new operational experience accumulated in IRS in the period 1995-2006, the project focuses on the identification of new CCI events. An attempt is also made to compare the observations made in the earlier study with the results of the current work, in particular regarding the potential dependency mechanisms and plant systems/areas that are most susceptible to dependent failures.

The main issues addressed within the scope of this project include:

- Extending the preliminary list of CCI event candidates generated in the previous pilot study [1, 2];
- Gathering additional experience in searching the IRS database for CCIs and formulating further guidance on CCI search strategy;
- Categorization of the identified CCI candidates;
- Comparison of the CCI candidates with EPRI list of Initiating Events;
- Identification of CCI groups of high concern;
- Providing further guidance on the scope and direction of further work in the area of CCIs.

1 General approach to event database review

1.1 Basic definitions

The following definition is used in this report for Common Cause Initiator:

Common Cause Initiator (CCI)

Common Cause Initiator is an event that causes simultaneous (or consequential) occurrence of an Initiating Event (IE) and functionally degrade or disable system(s) that are designed to cope with this initiator (mitigation systems).

Several elements of this definition deserve further explanation in the context of probabilistic safety assessment. They are discussed below.

Initiating event is a postulated event that creates a disturbance in a plant requiring some form of controlling or mitigation action, either manual or automatic. Such disturbances always lead to a perturbation in the heat production-removal balance of the plant and, depending on the successful operation or failure of various mitigation systems, have potential to lead to core damage.

It is worth noting that IE applicable to at-power plant operational states involve reactor trip. This is not the case for IEs that occur during shutdown. In the latter case the IEs typically involve disturbances affecting directly or indirectly the essential safety functions needed at shutdown such as the loss of coolant inventory, loss of cooling system, or loss of support to relevant systems.

Mitigation system is a plant system involved in providing a safety function that is required to cope with an initiating event. The mitigation system is closely associated with a specific IE under consideration. It is worth noting that mitigation systems credited in PSAs are not limited to safety systems; in many cases they include safety related systems or normal operation systems and support systems required for successful operation of these systems.

The concept of IE and mitigation system is closely associated with the event tree (ET) methodology. IE is the first element of an accident sequence definition followed by events related to success or failure of the required safety functions (functional ET) or of the related mitigation systems (systemic ET).

IEs originate from random failures of plant hardware (internal IE) or failures induced by hazards (internal or external). Therefore, they are always associated with a change in the hardware state of the plant. In a PSA the plant status determined by an IE is usually explicitly reflected in the related ET/FT plant logic model. In this approach the IE and the related logic elements of the ET/FT model are treated as independent events. Non-revealed dependencies between an IE and the related plant logic model elements will not be treated correctly and may lead to a considerable underestimation of the risk. That is the reason why CCIs are important and should not be overlooked.

1.2 Attributes of the database search

The definition of CCI discussed in Section 2.1 determines the basic attributes of the events that should be looked for in the event database. They include the following characteristics of the event:

(1) Effect of the event on plant operation;

(2) Degradation of safety significant SSCs;

(3) Failure type/mode of safety significant SSCs.

Effect of the event. CCI candidates should involve an initiating event. For events that occur at power the reactor trip is a necessary attribute since a sequence of occurrences initiated by any IE considered in PSA (including ATWS sequences) will finally end up with the reactor trip (automatic or manual). For IEs that occur at shutdown this is not the case (see Section 2.1).

Degradation of safety significant SSCs. In addition to an IE, one or more occurrences leading to functional degradation of the systems significant to safety should occur. The list of such systems should be narrowed to include only the systems involved in the mitigation of the specific IE (as credited in a PSA).

Failure type/mode. Dependency between the IE and at least one of the occurrences that causes a degradation of the mitigation systems associated with the IE is a necessary attribute of CCI events.

It is worth noting that the dependency between an IE and consequential degradation of relevant mitigation systems may be of two types - (i) direct or (ii) indirect.

Example of the first type is a RCS LOCA, which is an IE, and which due to its direct impact may degrade ECCS system that is needed for mitigation of LOCA accident.

Representatives of the second type are the situations/events in which one occurrence (failure of hardware or human error that is not IE itself) induces an IE and has a direct impact on the required mitigation system(s). Situations/events that involve internal hazards (fire and flood) are typical representatives of this type.

1.3 Events of interest in the database search

According to the definition of CCI (Sections 2.1) a CCI event needs to have three relevant attributes: (i) involving an IE, (ii) leading to a degradation of mitigation systems relevant for this IE, and (iii) occurrences that cause the degradation of the relevant mitigation system(s) being dependent on the IE.

Incidents reported in the IRS database are typically complex events that involve initiating event and degradation of several SSCs. Since by definition the IRS is focused on safety significant events/conditions only such events are selected for reporting in the IRS. However, the majority of events included in the IRS database are those that involve IE and several occurrences that cause degradation of mitigation systems but are not dependent on the IE. They have the first two attributes mentioned in Section 2.2.

Typically, the occurrences involved in these complex events (incidents) are either hardware failures that occurred before the IE (in standby systems) or are random faults (i.e. independent of the IE) that occur after the IE (e.g. failures of hardware to operate on demand or human errors in response to accident sequence originated by an IE). These complex events are considered as precursors of accident sequences leading to significant consequences (**Accident Sequence Precursors**).

It needs to be noted that certain number of IRS reports address issues that do not involve any IE. Typically, these reports describe degraded plant conditions identified during maintenance, testing or inspection. Some of the reports are dedicated to generic safety issues. Usually these reports are derived based on experience from several plants or address several events of similar type.

The review of IRS database to be conducted within the Project is intended to include investigation of all reports that may provide useful insights related to CCIs. It should be noted that the best source of information on the CCIs are real operational events of this type. However, as shown by the previous SKI project [1, 2], real Common Cause Initiators that are subject of interest in this Project are only a relatively small fraction of incidents reported in the IRS database. These events will be the main focus of the review. However, the review should also address other events (or issues) reported in the IRS database that could provide some useful insights related to the subject.

The IRS review performed within this study included also the incident reports that provide information on **potential CCIs** or those that address only a specific aspect related to the subject, e.g. **potential dependency mechanism** or **common cause failure mechanism** within the same system or a group of identical components. Further explanations and discussions of these technical terms are provided below.

Potential CCIs (P CCIs)

Potential CCIs are the events that involve IE and consequential degradation of some systems (dependent on the IE). However, the degraded systems are not relevant from the point of view of accident mitigation for this specific IE. Nevertheless, such events may disclose strong dependencies that in other conditions (or in plants with different design features) may lead to a degradation of mitigation systems. These events may be considered as **potential CCI events**.

Potential dependency mechanism (PDM)

Events that cannot be considered CCIs or potential CCIs but nevertheless involve occurrences with significant dependencies are also addressed in this study. The related IRS reports may provide insights on potential dependencies which in other plant conditions may originate a CCI. Typical representative of this group are (i) events that involve dependent occurrences but not led to IE, and (ii) degraded plant conditions that have a potential to lead to dependent occurrences causing an IE and at the same time degrading relevant mitigation systems.

Common Cause Failures

Events that involve dependent failures within the same system or a group of identical components (Common cause Failures) are also considered an additional source of information on the CCIs. In this study IRS reports that address CCFs or potential CCFs are analysed with regard to related coupling factor. It is expected that in some cases this may lead to identification of dependencies of new type that in specific plant conditions may originate a CCI.

It is worth noting that CCFs are modelled in PSA, to represent 'residual' dependencies (due to non-revealed causes) that cannot be modelled in a direct way. Typically, this model is applied to identical components. Dependencies addressed in CCFs belong to a broader group as compared to those considered in this study as PDM type. Potential coupling factors covered in CCF model include design, fabrication, maintenance, etc., i.e. factors that are not expected to have significant impact on multiple systems. Originating events of CCI type due to these dependencies is less likely.

2 Techniques used in database search

2.1 Basic concept

CCI attributes described in Section 2.2 provide a framework for establishing a basic concept of event database search. Coding available in the IRS database makes it possible to perform an automatic (computerized) search with regard to the first two attributes i.e. presence of IE (attribute 1) and the degradation of plant system (attribute 2). However, the capabilities of automatic search are limited with regard to the CCI identification.

Presence of the reactor scram that is the required attribute for IE for at-power operational states (attribute 1) can be addressed by automatic search of the database. IE applicable to shutdown operational states cannot be identified in this way; in this case more detailed analysis of degraded plant systems (attribute 2) needs to be performed. The required input for this evaluation is event description. Code based automatic search for a degradation of safety significant items (SSCs) may help in reducing the number of events subject to detailed investigation.

Providing information on the type of the IE involved using an automatic search is very limited. Search for the type of degraded systems/equipment (attribute 2) is not capable to narrow the list of systems to those involved in the mitigation of the specific IE. Investigations of this type have to be carried out manually based on event description (abstracts or full reports, if needed).

Possibilities to apply an automatic search for the type/mode of failure (attribute 3) are also very limited. A code-based search is not possible. A text-based search for key words may be helpful to identify presence of common cause/mode failures but capabilities of this search are limited in revealing the more sophisticated dependencies.

IRS database search implemented for the purpose of this project included several steps. Two reviews, one using the IRS coding system and another using a key-word search in the abstract description of the events were applied. The events identified in these two steps were investigated manually based on event descriptions (reading the abstracts or full reports).

Basically, the approach to database search was similar to that applied in the earlier study performed for SKI in 1998 [1, 2]. However, more attention was given to detailed analysis of events based on information provided in abstracts or full reports. Screening of events based on the event titles alone was reduced to a minimum.

2.2 Code-based search

Two code-based automatic searches were carried out as a first step of the database analysis/screening. In the first search all events with the reactor scram were selected. The second step was an independent search of the whole database for events that involved at least one failure or degradation of safety-related systems. Results of these two searches were combined. All new events identified in the second step were added to the event list generated in the first step, so the generated set of items included events with either the scram or the degradation of essential safety-related systems, or both.

The first step of code-based search was carried out using the two codes (related to the effect of event on operation):

- Code 6.1.1 (Automatic reactor scram) and
- Code 6.1.2 (Manual reactor scram).

The second step was intended to identify the degradation of plant ('failed/affected systems'). It addressed plant systems important to safety and selected support systems. The codes used in the second step included:

- Items important to safety (code 1.2),
- Essential reactor auxiliary systems (code 3.B),
- Essential service systems (code 3.C),
- Electrical systems (code 3.E),
- HVAC systems (code 3.H),
- Service auxiliary systems (code 3.K).

2.3 Key word search

Key-word search in the abstracts of the reports was intended to identify events that involve occurrences with potential dependencies. The key-words selected focus on identification of dependent occurrences/failures and on the items/system that are likely to be associated with dependent failures. The following key-words were used:

- common mode,
- common cause,
- potential to affect,
- multiple safety systems,
- multiple trains,
- clogged,
- pipe whip,
- instrument* failure,
- instrument* drift,
- drift,
- strainer,
- ventilation,
- service water,
- auxiliary feedwater,
- service water,
- power supply,
- AC power.

The new events (reports) identified in this step were screened out based on event title in order to eliminate events which are irrelevant from the point of view of CCIs. The remaining events were added to the list of events selected in the two code-based searches. This list was subject to a detail review based on the description of event provided in abstract or full report, if needed.

2.4 Manual search

Manual search based on event description was applied to events identified through automatic search (code-based and key-word searches) to eliminate events that were not CCI event candidates. Elimination of events in this step required a more careful analysis of each event based on event description. Practically, all items identified in the code-based search and selected events generated by the key-word search were subject to in-depth evaluation using information provided in the event description (abstract or full report, if needed).

Analysis included identification of direct cause of each event and relevant occurrences involved including an initiating event. Systems involved in the event (and their degradation) were identified. Events had to be scrutinized very carefully in order to identify elements relevant from CCI point of view. Important element of this analysis was the identification of dependencies between the IE and other occurrences. In most cases it was a difficult and time-consuming task.

3 Results of the review

Total number of reports in the IRS database at November 15, 2006 (the date of conducting the IRS search for the purpose of this study) was 3348. This number includes 2461 events with the date of the incident in the period of 01.01.1980 – 31.12.1995 that were addressed in the earlier SKI study [1, 2]. The current project addressed 899 IRS reports that correspond to the incidents that occurred in the IRS between 01.01.1995 and 15.11.2006[2].

It needs to be noted that the number of items (reports) recorded in the IRS database does not correspond to the number of operational events addressed in these reports. There are two reasons for this difference: (i) the database contains both preliminary and main (final) reports, and (ii) some of the reports address more than one operational event[3].

3.1 Overview of the search results

3.1.1 Initial search

The initial automatic search (code based and key-word searches) was applied to the whole set of database reports included in the review within this project (i.e. the 1995-2006 events sample containing 899 reports).

Table 3-1 provides information on the number of reports identified in each of the 3 queries. Queries A and B were fully automatic searches based on IRS codes. Query C was based on key-word search combined with manual screening (mostly using information provided by event titles).

Table 3-1. Summary overview of the database search results

Search	Search type	Number of items identified	Items selected for in-depth analysis
Query A	Codes 6.1.1 and 6.1.2	185 (a)	41
Query B	Codes 1.2, 3.B, 3.C, 3.E, 3.H, 3.K (c)	77 (b)	36
Query C	Key words (see Section 3.3) + manual screening (c)	34	25
Total		296	102

Notes:

a) Corresponding number of incidents is 183 (some events described in more than 1 report)
b) Corresponding number of incidents is 74 (some events described in more than 1 report)
c) Repeated items (i.e. those identified in earlier queries) are not counted.

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2 The year 1995 is an overlapping period in which there were 12 reports that are addressed both in 1998 project and in the current project.

3 This applies in particular to generic reports (e.g. NRC Information Notices) that address several events of similar type, which often occurred at different plants.

Out of the total number of 296 events identified using automatic search 102 items were selected for further analysis. Based on the event description these events were considered candidates for one of the CCI-related categories described in Section 2.3. The remaining 194 events were considered out of interest from the point of view of CCIs. They were mostly simple initiators or accident precursors that did not involve any dependent failures.

Appendix A (Tables A-1 – A-3) provides information on the 296 events selected based on automatic searches (queries A – C). This includes basic information on the events taken from the IRS database records (such as IRS report number and plant type), and a summary of information derived from the analysis of event abstracts and/or full reports. A brief description of each of the events is provided with focus on CCI-related aspects, and the type of initiating event, if applicable. IE type is assigned using the classification provided in the EPRI generic list of IEs (for PWR and BWR plants). Finally, based on the event characteristics an indication is given on items which seem to be of interest from the point of view of CCIs.

3.1.2 In-depth analysis

The selected set of events (102) was subject to in-depth analysis that concentrated on the type of IE (if applicable), direct cause of the event, additional systems degraded during the event, and type of dependencies involved. Based on this information the events were divided into CCI-related categories described in Section 2.3.

Out of this set 11 events were identified as generic safety issues that do not bring any potential insights regarding the CCI subject. These events were screened out from further analysis (in Appendix A these items are marked by "*" in the last column).

The remaining 91 items are candidates for CCIs or events that provide useful insights regarding potential dependency mechanisms. They were assigned into one of the following categories:

- Common Cause Initiators (CCIs)
- Potential Common Cause Initiators (P CCIs)
- Events that indicate potential dependency mechanism (PDM), and
- Common Cause Failures (CCFs)

Appendix B provides CCI-related information for this set of events. This includes the event type, direct cause, degraded systems, event category (CCI, P CCI, PDM, CCF, as discussed in Section 2.3), and a brief description of the nature of dependencies involved.

It should be noted that information provided in Appendix B regarding the 'direct cause' focuses on occurrences that are considered crucial from the point of view of the involved dependencies between IE (potential IE) and degradation (potential degradation) of the 'mitigation systems' associated with this IE (see discussion in Section 2.2).

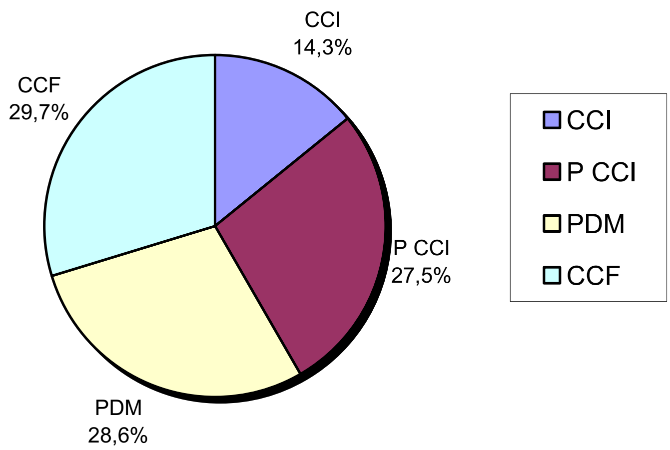


Figure 3-1. *Events of interest identified in the IRS database in relation to CCIs.*

3.1.3 Event categorization

The selected events were categorized with regard to the 'direct cause' of the event and the 'dependency mechanism'.

Direct cause

It needs to be noted that the analysis of the direct cause of the event focuses on occurrences that would trigger a chain of dependent occurrences leading to IE and plant degradation (mitigation systems). In some cases (when IRS report addresses degraded plant conditions rather than real events) these occurrences are 'postulated events' that would originate dependent failures. Appropriate information on this subject is provided in Appendix B.

Attributes used in the categorization of 'direct cause' occurrences include the malfunction type (human action or hardware failure) and the type of system/component involved.

The human interactions included:

- Test related errors
- Operational error
- Maintenance error
- Design/fabrication error

Hardware (component/system) failures included types of SSCs:

- Mechanical systems/components
- Electrical systems/components
- I&C systems/components.

Fig. 3-2 shows a distribution of 'direct cause' occurrences among the three types of systems and contribution of hardware failures and human errors (and their combination, if applicable). This figure addresses only three groups of events: CCI, P CCI, and PDM. The CCF group was excluded since the related IRS reports address mostly degraded plant conditions (generic issues) for which the 'direct cause' concept is hardly applicable.

Fig. 3-3 shows a distribution of 'direct cause' occurrences in terms of component types for each of the system categories.

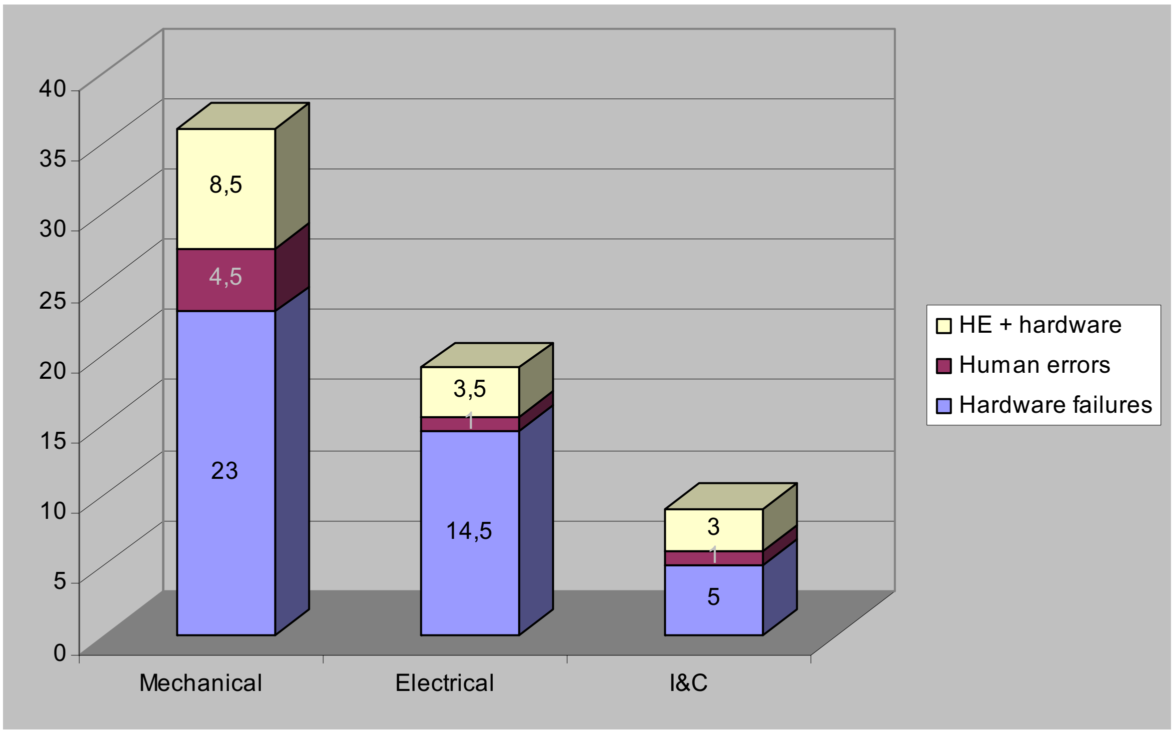


Figure 3-2. *Systems and types of malfunctions involved in the direct cause of events (64 events of CCI, P CCI, PDM type reported to occur in the period of 1995-2006)*

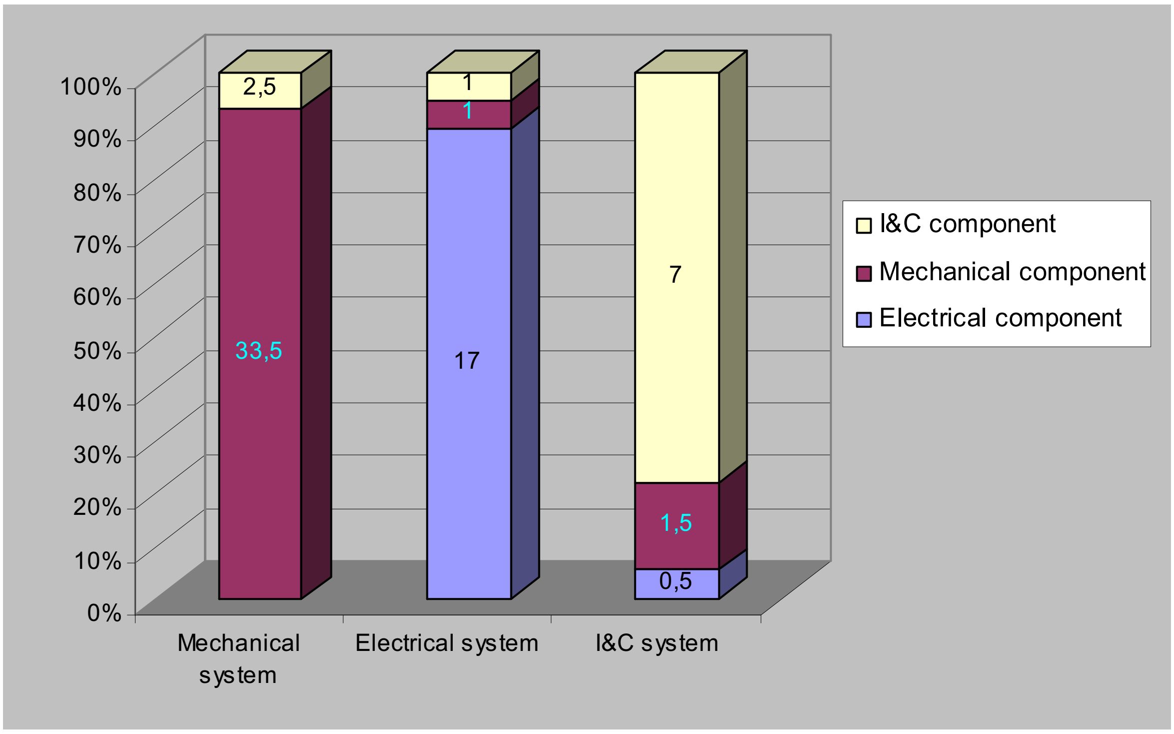


Figure 3-3. *Components involved in the direct cause of events for system groups (64 events of CCI, P CCI, PDM type reported to occur in the period of 1995-2006)*

As shown in Fig 3-2 the distribution of events among system type categories is not uniform. About 56 % of direct cause occurrences involve the mechanical systems. Electrical systems contribute ~30 % and I&C systems 14 %. Contribution of hardware failures varies depending on the system type from ~76 % for the electrical systems to ~55 % for the I&C systems. The remaining contributors are human errors alone or in combination with hardware failures. The latter group includes also HEs made before the incident. As shown in Fig 3-3 failures of systems are dominated by failures of component of the same type – the mechanical systems by failures of mechanical components, the electrical systems by failures of electrical components, etc.

It is worth noting that the value of quantitative assessment regarding the direct cause is limited. There are several factors contributing to this limitation.

One of the factors is the lack of consistent principles/criteria for classification systems and components involved in direct cause. For instance, in case of an event initiated by a malfunction of a valve (mechanical component) in a system of mechanical type, the analyst may be tempted to classify the related direct cause as M/M (mechanical component/mechanical system). However, if the evaluation goes deeper it may appear that the problem occurred due to a failure of related I&C component (I) or electrical component (E). In this case the classification could be I/M or E/M. This introduces some subjectivity in the classification.

Another difficulty is that in some cases there is a combination of several failures (components of different types) or even malfunctions in more than one system. Additional complication is the assignment of the involved system in case of events initiated by human error.

Additional problem is that the items addressed/counted in the evaluation are not the individual events but the IRS reports. The NRC generic reports are typical representatives of such cases. Typically, reports of this type are related to several similar events but with different direct cause and different systems/components involved. In some generic reports there is no sufficient information on the number of events. Therefore, in the current project the statistics is based on the number of items (reports) included in the IRS database. Exception from this rule is that the preliminary and the main reports are considered as one ‘item’.

Dependency mechanism

Categorization of events with regard to ‘dependency mechanism’ was conducted for all events considered interesting from CCI point of view (91 events including CCI, P CCI, PDM and CCF type). Fig. 3-4 shows distribution of these events among the major categories:

- Support (power, ventilation, computer, air, etc.)
- Protection/control (signal, measurement, setpoints)
- Maintenance
- Direct /physical
- Common cause (residual)
- Environment/Area
- Transient (operational and accident-induced)
- External

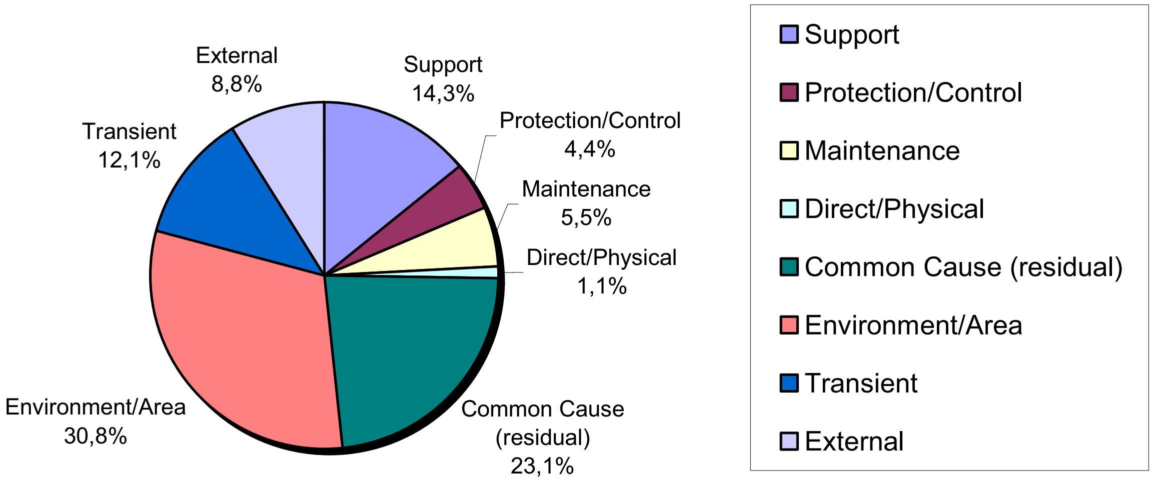


Figure 3-4. *Various dependency mechanisms involved in CCI-related events – major types (91 events reported to occur in the period of 1995-2006)*

More detailed categorization of the observed dependency mechanisms is shown in Figs 3-5 and 3-6.

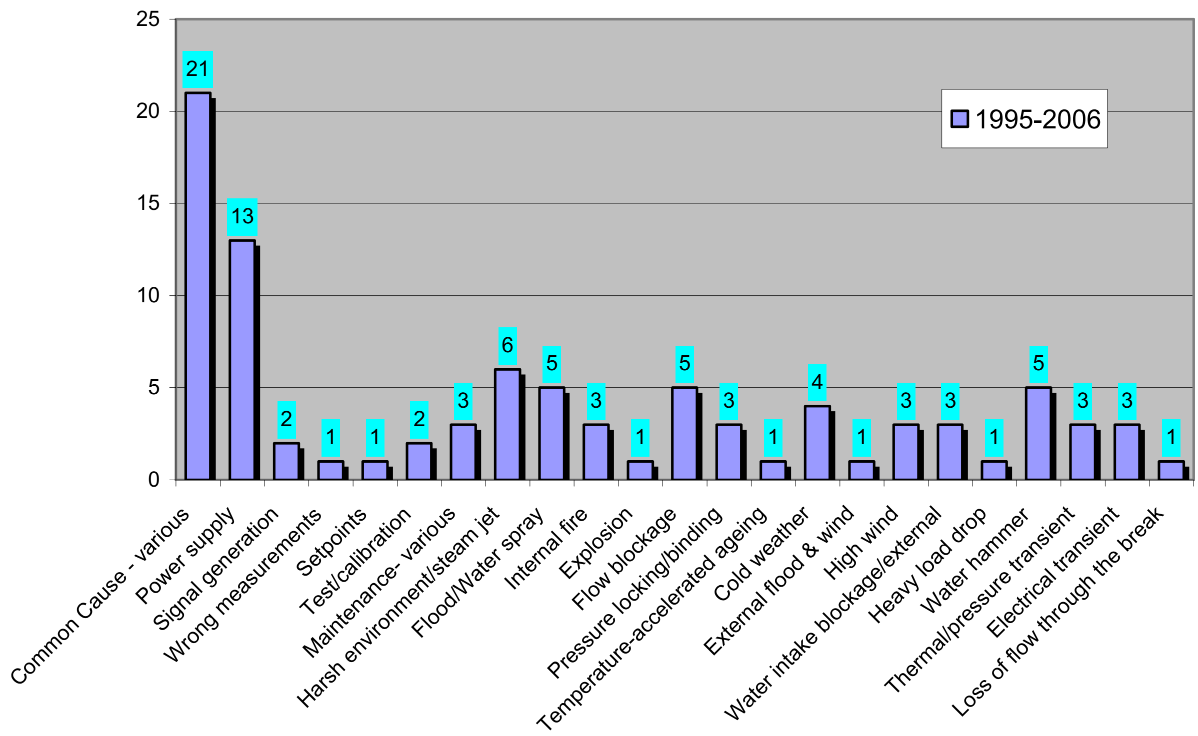


Figure 3-5. *Various dependency mechanisms involved in CCI-related events – detailed categorization (91 events reported to occur in the period of 1995-2006)*

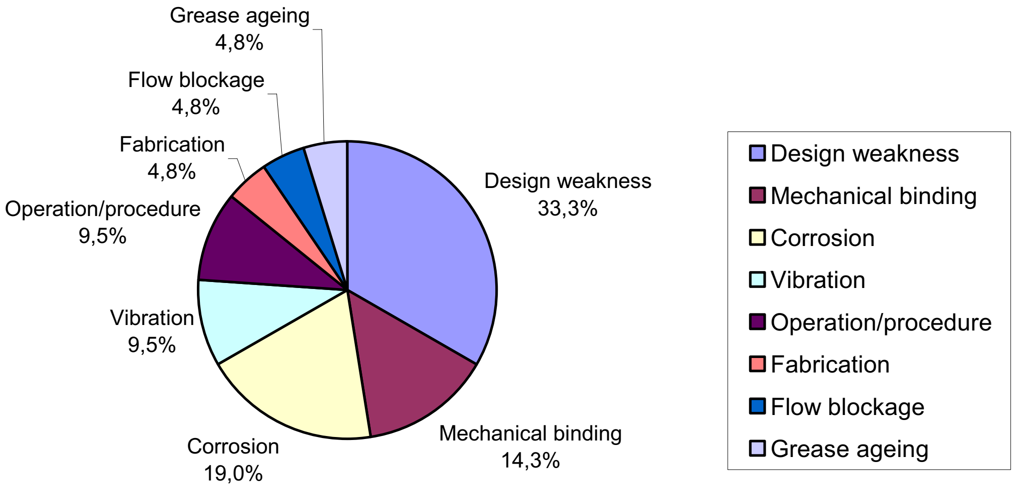


Figure 3-6. *Coupling factors involved in Common Cause Failure events (residual) – (summary of 21 IRS reports issued in the period of 1995-2006)*

The most relevant dependency mechanisms are related to environmental impact/area events, direct dependencies through support systems, and transients. Fig. 3-5 shows individual contributors in these major groups. The most frequent are power supply, harsh steam environment and steam jet impact, flood and water spray, flow blockage, water hammer, and cold weather.

In the group of CCF related events the 'design' is the most important coupling factor. Although this coupling factor is not likely to contribute to CCIs, there are other coupling factors that may be relevant from CCI point of view. It includes factors of environmental type (corrosion, vibration, mechanical binding).

It needs to be noted that the quantitative insights regarding the dependency mechanisms are limited. Discussion provided in Section 4.1.3.1 on the factors that have adverse impact on the value of statistics derived in this project is also applicable to the evaluation of dependency mechanisms.

3.1.4 Comparison with EPRI list of IEs

The assignment of IEs based on EPRI categorization [3] was conducted for each of the events selected as CCI candidates. The results of this task are documented in Appendix A.

Assignment of IEs to categories defined in EPRI IE list is not always straightforward since the events are often very complex chain of occurrences leading to highly degraded plant states.

General approach applied in this task was to select the most "simple" occurrence that led to a perturbation in the heat production-removal balance of the plant. Other occurrences were considered as elements of accident sequence progression. Therefore, the selected initiating events usually do not reflect the complex conditions associated with the event.

The EPRI list of IEs developed for PWRs and BWRs was also used in this project for other reactor types. In case of IRS reports that apply to more than one event the IE categorization was not performed.

4 Integration of results with 1998 SKI project

Results of the current project were compared and integrated with the results of the earlier CCI study [1, 2]. The most important observations from this process are discussed in the following sections.

4.1 Structure of CCI data sample

The first SKI project addressed 2461 IRS events that occurred in the time period 1980-1995. There were 60 events out of this set which were considered interesting from the point of view of CCIs. Together with 91 events selected in the current project the total number of events selected is 151. Structure of this sample is shown in Table 4-1 in terms of the major CCI categories.

Table 4-1. Structure of CCI-related events selected in the 1998 and 2006 projects.

CCI-related category	1980-1995		1995-2006		1980-2006	
	Events	%	Events	%	Events	%
CCI	23	38,3	13	14,3	36	23,8
P CCI	13	21,7	25	27,5	38	25,2
PDM	22	36,7	26	28,6	48	31,8
CCF	2	3,3	27	29,7	29	19,2

In the review conducted earlier [1, 2] the fraction of ‘real’ CCIs identified is higher and the fraction of potential CCIs lower as compared to the current project. The number of potential CCIs identified in the earlier study is lower as compared to the current project. The current review also identified a significantly larger amount of events considered as PDM or CCFs.

The overall yearly rate of CCI-related events (all categories) for the period 1980-95 is much lower (~4 per year) as compared to the period 1995-2006 (~7.6 per year). This difference is even higher when consideration is given to the total population of events contained in the IRS database for the period 1980-1995 and 1995-2006 (2461 vs. 899, respectively).

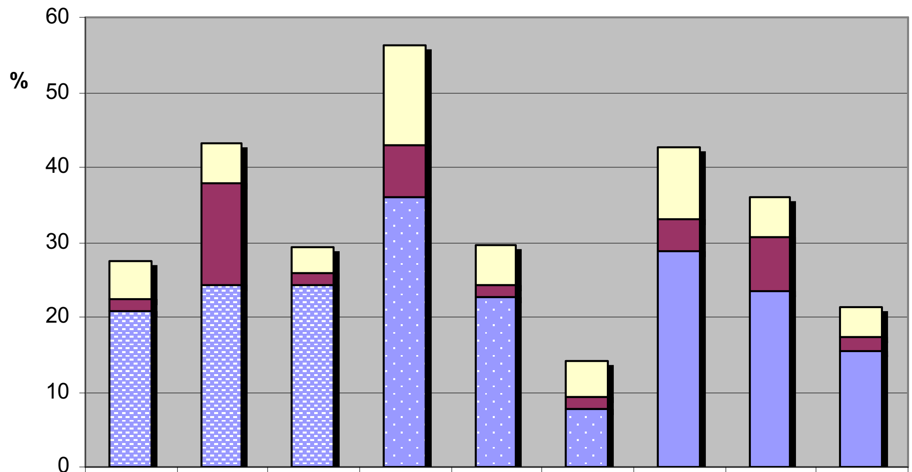
There are several factors that may contribute to these differences. One of the reasons is probably the change in the IRS reporting criteria/practices that evolved before the system reached the full maturity. In the early phase of IRS the number of ‘less significant’ events was probably higher. Additional factor is that in the later period there are more reports that address generic safety issues covering more than one operational event. One of the contributors to this effect can also be the fact that the events are reported in IRS with a significant delay. Certain events that had occurred in the period of 1995-2006 have not being reported yet.

4.2 Categorization aspects

4.2.1 Direct cause

Distributions of systems and types of malfunctions involved in the direct cause of events are shown in Fig. 4-1 for the three events samples - 1980-1995, 1995-2006, and 1980-2006.

As can be noticed, the systems contributing to the 'direct cause' are different in the current project as compared to the earlier results. In the period 1995-2006 contribution of systems of the mechanical type is higher than the electrical and I&C systems.



	1980-95			1995-06			1980-06		
	Mech.	Electr.	I&C	Mech.	Electr.	I&C	Mech.	Electr.	I&C
HE + hardware failures	5,2	5,2	3,4	13,3	5,5	4,7	9,4	5,3	4,1
Human errors	1,7	13,8	1,7	7,0	1,6	1,6	4,5	7,4	1,6
Hardware failures	20,7	24,1	24,1	35,9	22,7	7,8	28,7	23,4	15,6

***Figure 4-1**. Systems and types of malfunctions involved in the direct cause of events – relative (%) contribution (based on events of CCI, P CCI, PDM type i.e. 58 and 64 events in the period of 1980-1995 and 1995-2006, respectively)*

4.2.2 Dependency mechanism

Distributions of events in terms of various dependency mechanisms are shown in Figs 4-2 and 5-3 for the three events samples (1980-1995, 1995-2006, and 1980-2006).

The 3 most significant dependency categories identified in the earlier study (power supply, harsh environment, internal hazards (flood and fire), water hammer, and cold weather are also clearly noticeable in the current event sample.

In the 1995-2006 events sample the 'power supply' factor is the only contributor in the group of 'support systems'. Also the total fraction of items in the 'support' category is smaller as compared to the earlier 1980-1995 sample. Similar observation can be made regarding the contribution of internal flood and fire.

It can be noted that the lower contribution of the events that involve 'power supply' dependency mechanism observed in the current project (event sample 1995-2006) is one of the factors that result in a lower contribution of 'electrical' and 'I&C' systems involved in the direct cause (as shown in Fig. 4-1).

The current review identified the dependency mechanisms that were not observed in the earlier event sample (1980-1995) or contributed very little. These include the maintenance, flow blockage, pressure locking / binding, and external events (wind).

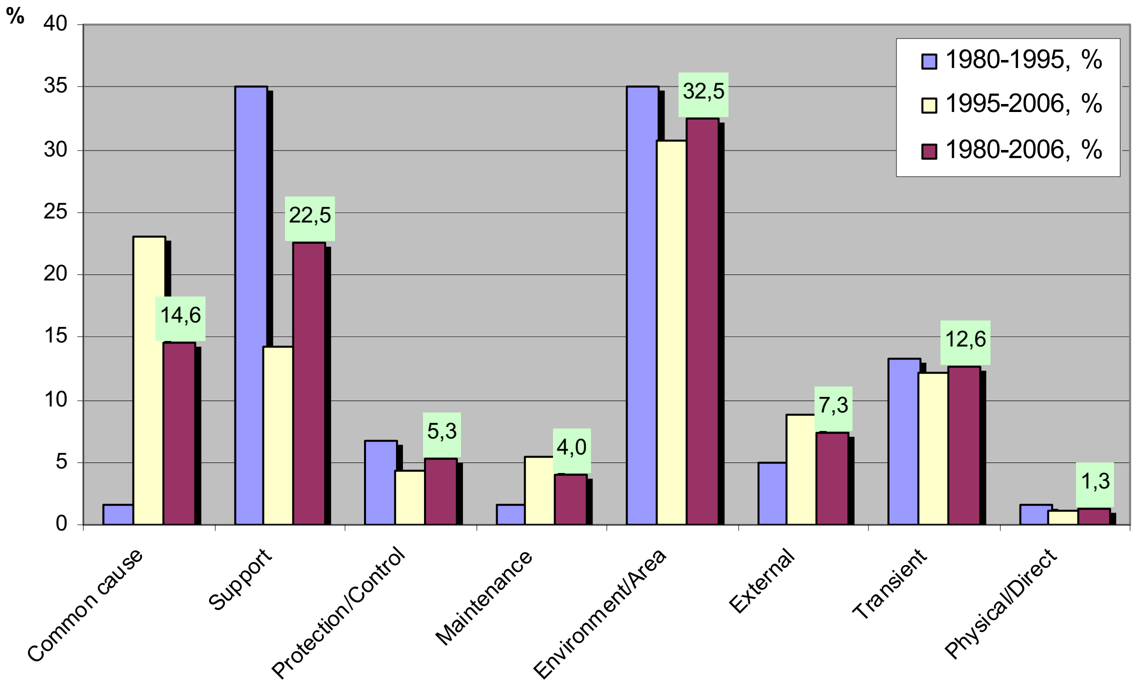


Figure 4-2. *Relative contribution (%) of various dependency mechanisms involved in CCI-related events – major categories; (all types of events (CCI, PCCI, PDM, CCF) i.e. 60 and 91 events in the period of 1980-1995 and 1995-2006, respectively).*

The dependency mechanisms related to maintenance, flow blockage (internal and external causes), and pressure locking in mechanical systems (e.g. valves) are addressed in several generic reports that refer to multiple events of this type. The same observation applies to the cold weather dependency mechanism. Therefore, the importance of these factors is even higher than it looks like based on the current statistics.

The maintenance issues (complex work situation, time pressure and incorrect procedures) are addressed in the generic report of Sweden (IRS 7303). It refers to 9 events in Swedish BWRs and PWRs. Maintenance problems that may originate dependent occurrences are also mentioned in generic IRS reports devoted to CCFs (IRS reports 7339 and 7436), and other reports not selected for in-depth evaluation in this project (e.g. IRS reports 7274 and 7499).

The flow blockage issue (internal – RHR pump strainers and ECCS sump screens, and external - water intake screens) is addressed in several generic reports (e.g. IRS reports 1546, 7600, 7626, 7656, and 7813).

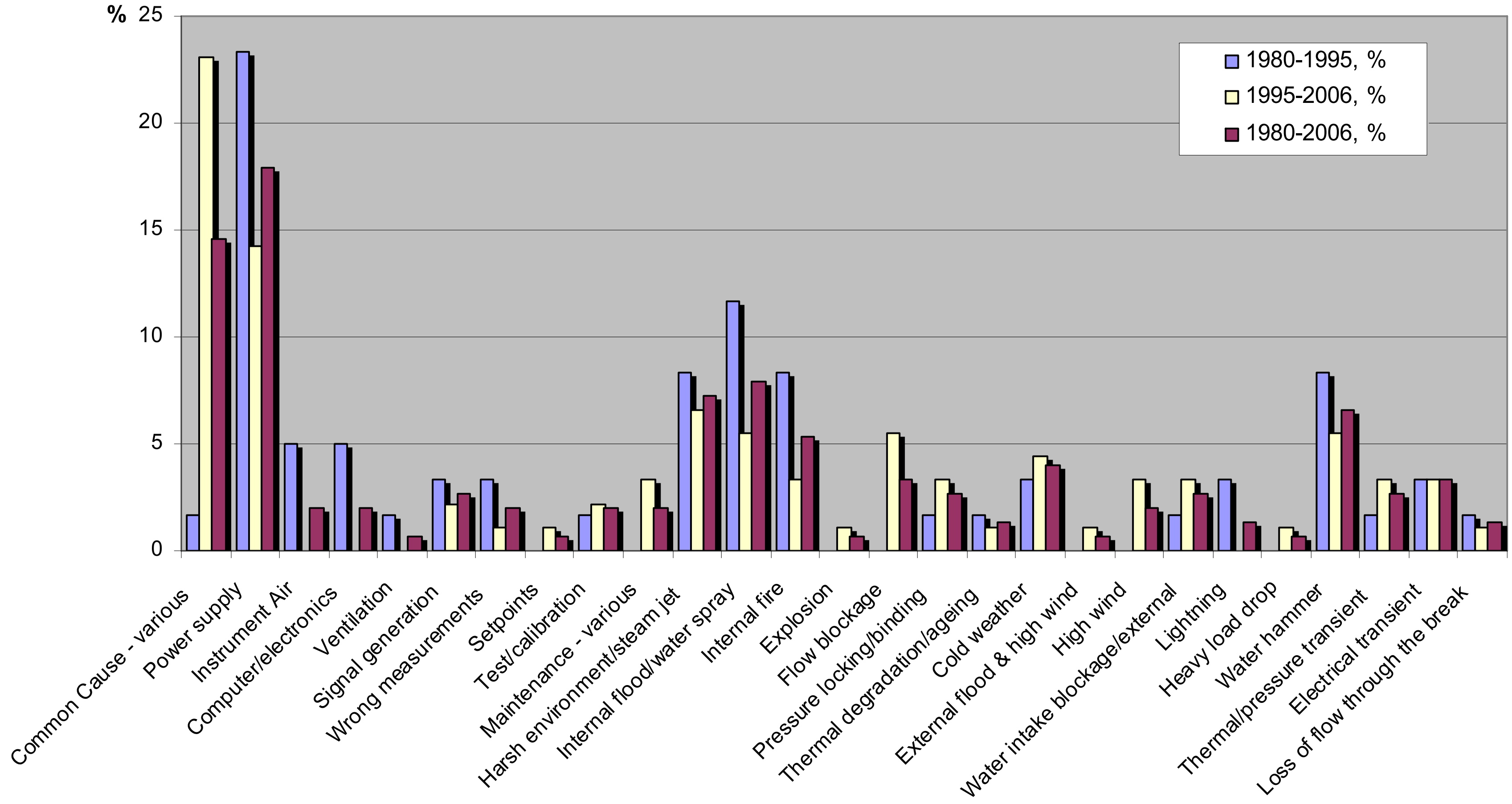


Figure 4-3. *Relative contribution (%) of various dependency mechanisms involved in CCI-related events –detailed categories (all types of events (CCI, PCCI, PDM, CCF) i.e. 60 and 91 events in the period of 1980-1995 and 1995-2006, respectively)*.

The pressure locking of mechanical components (ECCS and CSS valves) is considered a generic issue and discussed in generic reports issued by the NRC (IRS reports 1525, 1599, and 7135).

The issue of cold weather is addressed in several reports of generic type. One of the NRC generic reports devoted to this subject (IRS report 7180) noted 37 cold weather related events at 23 different sites between 1991 and 1997. The study also reported an increasing trend in the number of these events. Generic report issued by the Russian Federation issued in 1997 describes 4 operational events of this kind that occurred in the period 1987-1995.

5 Conclusions

The review of the NEA IRS database conducted within this project generated a sample of events that provides insights regarding the Common Cause Initiators (CCIs). This list includes certain number of 'real' CCIs but also potential CCIs and other events that provide insights on potential dependency mechanisms.

Relevant characteristics of the events were analysed in the context of CCIs. This evaluation was intended to investigate the importance of the CCI issue and also to provide technical insights that could help in the modelling the CCIs in PSAs.

The results of the project were compared and integrated with the results obtained in the earlier CCI study conducted in 1998 [1, 2].

5.1 General conclusions

The project confirmed general findings of the earlier project.

- Events of CCI type occur in operational practice of NPPs. Such operational events have been reported in the IRS database.
- Majority of operational events documented in the IRS database are complex events that include many occurrences both consequential and random. Most of these events are accident precursors. The events that involve consequential (dependent) degradation of the mitigation systems (the required attribute of CCI) are relatively rare events. In addition, the extent of this degradation is usually limited. Partial failures (degradation) of system functions are typical. Loss of automatic features or safety related indications are examples of such degradation.
- In addition to 'real' CCIs there are also potential CCI events which in other plant conditions may originate 'real' CCIs. Important source of information on the subject of CCIs, in addition to 'real' operational events of CCI type, are the potential CCIs as well as other operational events that involve dependent failures/malfunctions.
- The review performed within the framework of both projects was limited in scope. For some of the events a more systematic detailed analysis of the event would be needed in order to understand properly the course of events and relationship between the occurrences. Distinction between the real CCIs and potential CCIs is in some cases based on subjective judgement.
- Some engineering insights of quantitative nature (e.g. statistics on the contribution of systems involved in the direct cause, or on the dependency mechanisms) are of limited value. Factors contributing to this drawback include the weaknesses of the analysis performed within the Project as well as the inherent limitations of the IRS. However, in spite of these shortcomings the project provides useful qualitative insights that may help in better understanding of the subject of CCI and their origin (e.g. by identification of new dependency mechanisms).

5.2 CCI identification

The following observations were made regarding the process of CCI identification:

- Identification of CCI candidates is difficult and time consuming. Application of automatic search in the database is limited. Manual search/analysis based on in-depth evaluation of event characteristics is important element of the review.
- In-depth analysis of events performed within the current project required information that is normally provided in the full IRS reports. Information from IRS abstracts was in most cases insufficient.
- The identification of consequential failures was the most difficult and time-consuming part of the assessment which required access to detailed information on the event. In case of complex incidents, representing the event in terms of individual occurrences is an essential element of the CCI-focused analysis. Within the current project a systematic analysis of this type was limited. This fact has some impact on the level of uncertainties associated with categorization of events.

5.3 Engineering insights from the study

The analysis of operational events provided useful engineering insights regarding the potential dependencies that may originate CCIs. Some indications were also obtained on the plant SSCs/areas that are susceptible to common cause failures. The following observations have been made:

- Direct interrelations between the accident mitigation systems through common support systems, which can originate a CCI, represent a dominant dependency mechanism involved in the CCI events. This observation made in the earlier CCI study was confirmed in the current project.
- The most important contributors of this type are electrical power supply systems and I&C systems. One of the mechanisms contributing to the degradation of the in-house electrical power supply systems is associated with malfunctions or spurious operation of the equipment protection devices/systems. Evidence for interrelations of this type has been found in the existing operational experience (in the 1995-2006 events sample).
- Area-related events such as internal fires, flooding, water spray, and steam jet have also been found to be important sources of dependency. Contributions of these dependency mechanisms are slightly lower in the current events sample (1995-2006) as compared to the results of the earlier study, but distinctly visible.
- The current review shows that transients (e.g. water hammer, electrical transients both internal and external) as well as external events (e.g. lightning, high wind or cold weather) are relevant sources of dependency that may originate CCIs.
- Majority of the direct and area-related dependencies are plant specific and determined by plant design features. The analyst can identify them and model explicitly in a PSA. Modelling of these dependencies requires a detailed analysis of all system interrelations.
- Direct dependencies require a systematic analysis of failures in electrical distribution system (including DC power) and in the engineered safeguards. The FMEA may provide the required input to PSA modelling.
- The review of operational events shows that in some cases a single failure of I&C or electrical component combined with non-revealed deficiencies in other plant systems, not necessarily safety-related (e.g. equipment protection or load re-distribution/reinstating automatics) may originate a complex incident (including severe transient) that leads to a degradation of several plant SSCs. Examples of unexpected effects e.g. induced by blocking of impulse lines (due to condensation, gas bubble, or freezing) exist in the IRS (1995-2006 events sample).

- The methodology for area-related events, such as fire and flooding, is relatively well established. Operating experience shows that in the analysis of internal flood the effects of localised flood or spraying of electric components may be of high importance and deserve proper attention.
- The new dependency mechanism related to the flow blockage and pressure locking can also be modelled in PSA in a direct way. In some of the current PSAs the dependencies of this type are represented as CCFs.
- Transients (e.g. water hammer, electrical transients both internal and external) as well as external events (e.g. lightning, high wind or cold weather) are potential dependency mechanisms that have not been modelled in PSAs and that are more difficult to be considered explicitly in the plant logic model.
- Another dependency mechanism, difficult to be modelled, is related to human factors. This applies to non-conservative planning of maintenance (one of the issues identified in the project) and errors of commission.

REFERENCES

[1] Identification of Common Cause Initiators in IRS Database, SKI Report 1998:9, February 1998

[2] Identification of the CCI and their precursors documented in the NEA Incident Reporting System, SKI/RA-023/97

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Defining initiating events for purposes of probabilistic safety assessment, IAEA TECDOC-719 (1993) (pages 28 and 41).

6 Appendix A

Summary of detailed analysis for the events selected using automatic search (Queries A, B, C)

Table A-1. *Query A search results (Search codes: 6.1.1, 6.1.2; Incidents between 1995-01-01 and 2006-11-15; total of reports=185)*

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A1	1469	MANUAL REACTOR SCRAM AT LAGUNA VERDE NUCLEAR POWER PLANT UNIT 1 BECAUSE OF POWER OSCILLATIONS	Preliminary report			
A2	1469	MANUAL REACTOR SCRAM AT LAGUNA VERDE NPP UNIT 1 BECAUSE OF POWER OSCILLATIONS	(BWR) Power oscillations outside the region of instability in the core (BWR); manual scram	BWR	BWR(36)	
A3	1492	HEAT TRANSPORT SYSTEM LIQUID RELIEF VALVE FAILURE	(PHWR) Reactor trip on low PRZ level. Relief valve in the primary pressure and volume control system failed open (valve actuator failure). Failure or significant degradation of heat removal capability. Safety valves of the bleed condenser opened causing spill of water to the containment.	PHWR	PWR(7)	
A4	1496	DAMAGE TO THE MAIN TRANSFORMER DUE TO AN INTERNAL VOLTAGE OSCILLATION	(BWR) Line-to-ground fault in the transmission line due to a lightning strike. Turbogenerator trip and reactor trip.	BWR	BWR(3)	
A5	1511	STEAM LEAK ON LIVE STEAM INLET TO TG/2 MOISTURE SEPARATOR /REHEATER	(PWR) Disturbance in the secondary system and steam leakage (Moisture Separator / Reheater). Transient with loss of MFW	PWR	PWR(28)	
A6	1546	UNEXPECTED CLOGGING OF A RESIDUAL HEAT REMOVAL (RHR) PUMP STRAINER WHILE OPERATING IN SUPPRESSION POOL COOLING MODE (NRC BULLETIN 95-02)	(BWR) Safety Relief Valve inadvertently opened (5 SRVs leaking before IE). RHR operating in the suppression pool cooling mode. CCM blockage of ECCS strainers. Sludge (iron oxide) and fibers (polymer) blocking, inadequate pool cleanliness.	BWR	BWR(11)	YES
A7	1608	REACTOR AUTOMATIC SHUTDOWN CAUSED BY ACTUATION OF STEP-UP TRANSFORMER PROTECTIVE RELAY	(PWR) Annual inspection work, Due to human error (wrong relay isolated) generator and reactor trip	PWR	PWR(34)	
A8	1610	SLOW FIVE PERCENT SCRAM INSERTION TIMES CAUSED BY VITON DIAPHRAGMS IN SCRAM SOLENOID PILOT VALVES (NRC INFORMATION NOTICE 96-07)	(PWR) Scram insertion time too long due to a problem with diaphragms in scram solenoid pilot valves (generic issue). In Vermont Yankee Dec 8, 1995 a scram due to FW oscillation slower rate of insertion observed	PWR	PWR(21)	YES
A9	1633	EXCESSIVE COOLDOWN OF PRIMARY SIDE TO MAINTAIN SUFFICIENT SUBCOOLING MARGIN DURING UNEXPECTED PRESSURIZER SPRAY VALVE OPENING	(PWR) Pressure transient induced due to PRZ response to false signal in PRZ pressure control channel. PRZ spray valves and relief valve opened. Spray valve stuck open (mechanical failure), plant trip, system cooled down excessively by the operators (independent of previous failure)	PWR	PWR(7)	
A10	1634	SAFETY INJECTION BY TURBINE DIGITAL ELECTROHYDRAULIC CONTROL CARD FAILURE	(PWR) Replacement of faulty card in turbine protection system, false turbine overspeed signal, response to false signal induced transient in steam system and initiated safety injection on low pressure in the MSC	PWR	PWR(9)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A11	1635	D2O SPILLAGE THROUGH AUTOCLAVE SYSTEM DRAIN LINE	(PHWR) Internal leak from the primary system through failed drain isolation valve on the autoclave system	PHWR	PWR(5)	
A12	6396	OPERATION OF REACTOR PROTECTION SYSTEM DUE TO OPERATOR ERRORS WHILE ADJUSTING THE PROTECTION SETTINGS	(PWR) HE in resetting setpoints in RPS leads to reactor trip.	PWR	PWR(39)	
A13	6406	REACTOR SCRAM DUE TO A SPURIOUS OVERPOWER SIGNAL	(PWR) Reactor transient due to spurious signal caused by HE during testing that led to manual trip of TG and reactor	PWR	PWR(40)	
A14	6409	SAFETY INJECTION ACTUATION AND PLANT SHUTDOWN	(PWR) Defected fuse caused MSIV closure and transient with initiation of ECCS injection and reactor trip	PWR	PWR(18)	
A15	7000	EXPLOSION OF A CIRCUIT BREAKER	(PWR) Non-safety related system (Condenser /circulating water system affected). No effect on plant safety, manual reactor trip	PWR	PWR(40)	
A16	7002	REACTOR TRIP ON LOW PHT PRESSURE DUE TO CHANGE IN POWER WHILE REVERSE FUELLING OF CHANNEL.	(PHWR) Power increase to 109% due to disturbance in on-line refuelling (‘reverse refuelling – against the direction of flow) led to reactor trip	PHWR	PWR(40)	
A17	7003	NON CLOSURE OF PRIMARY STEAM ISOLATING VALVES (PSIV's) ON AUTO DURING TOTAL LOSS OF POWER INCIDENT.	(BWR) Loss of off-site grid led to reactor trip, problem with one diesel and primary steam isolation valves (failed to close as required in this case), they were closed by operators (probably rust in pilot valve)	BWR	BWR(31)	
A18	7008	REACTOR TRIP BY THE SIGNAL OF CONTROL ELEMENT ASSEMBLY DEVIATION	(PWR) Trip due to failure in RPS (false DNBR signal)	PWR	PWR(39)	
A19	7009	THE REACTOR SHUTDOWN BY STEAM GENERATOR HI-HI LEVEL	(PWR) Malfunction of the FW control circuit led to abnormal opening of the FW control valve and to high level in SG, turbine and reactor trip followed	PWR	PWR(19)	
A20	7011	PLANNED REACTOR SHUTDOWN DUE TO STUCK GUIDE SLEEVE OF FUELLING MACHINE	(PHWR) Manual shutdown of the reactor due to problem with refuelling machine	PHWR	PWR(40)	
A21	7014	MAIN STEAM ISOLATION VALVE STEM FAILURE	(PWR) MSIV inadvertent closure (mech. failure of the valve stem) led to reactor trip, all systems available	PWR	PWR(18)	
A22	7032	REACTOR SCRAM DUE TO STEAM GENERATOR LEVEL SIGNAL DROP FOLLOWING NaOH INGRESS INTO TURBINE CONDENSATE LINE	(PWR) Ingress of NaOH to the condensate tanks (HE during maintenance of filter) foam was produced in 3 SGs that caused SG level low signal, transient and manual trip of both TGs	PWR	PWR(40)	
A23	7050	DEGRADATION OF COOLING WATER SYSTEMS DUE TO ICING (NRC INFORMATION NOTICE 96-36)	Generic issue based on 3 events from the past. Icing conditions caused degraded operability of some systems (circulating, service and fire water) and reactor trip (manual)		PWR(40)	YES

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A24	7052	UNEXPECTED OPENING OF MULTIPLE SAFETY RELIEF VALVES (NRC INFORMATION NOTICE 96-42)	(BWR) Multiple relief safety valves opened without apparent initiating cause. Operators scramed the plant manually to prevent excessive temp in the suppression pool. The root cause identified was a transient in the 24 V DC in SRV logic module	BWR	BWR(11)	YES
A25	7054	LOSS OF OFF-SITE POWER SOURCES OF BILIBINO NPP DURING RMG TESTING	(LWGR) Testing reversible motor generator in electrical system during scheduled maintenance led to loss of essential buses in the units. Due to additional HEs there was a loss of off-site power to twin units that were under operation.	LWGR	BWR(31)	
A26	7056	PARTIAL FAILURES OF ICMS AND LOSS OF CONTROL ROD POSITION INDICATION DUE TO CONTAINMENT SPRAY ACTIVATION	(WWER) Testing of spray system led to release of boric acid to the containment and failure of the In-Core Monitoring System (ICMS) and CR position indication (contacts affected by boric acid). Manual shutdown to remove problem. (incorrect adjustment of end switches of a valve at the SS train discharge.	WWER	PWR(40)	
A27	7057	NPP OPERATIONAL EVENTS RELATED TO ABNORMAL IMPACT OF ENVIRONMENT ON EQUIPMENT OPERATION IN THE COLD TIME OF THE YEAR	WWER generic (4 plants). Combination of cold conditions with human errors (mngt and org) led to transients. Freezing impulse lines in secondary system (SG, MSC) led to spurious signals and malfunctions of components	WWER		YES
A28	7062	A MISACTUATED SI DUE TO DISPOSITION OF A BYPASS SWITCH DURING SURVEILLANCE TESTING	(PWR) RPS monthly test + HE (mis-positioned bypass switch) led to false signal initiation of safety injection and reactor scram.	PWR	PWR(9)	
A29	7070	TRIP OF REACTOR 22 DURING AUTO CONTROLLED POWER RAISE FOLLOWING ZONE ROD CONTROL FUSE FAILURES	(GCR) Failed fuse in the control rod drive motor + wrong operator action confused by the problem led to reactor trip (during reactor start-up)	GCR	PWR(39)	
A30	7072	TRIP OF 1 AND 2 STATION TRANSFORMERS AND SUBSEQUENT DOUBLE REACTOR SHUTDOWN	(GCR) Low oil level in station transformer, change to auxiliary power supply, followed by a trip of both station transformers. Trip of Reactor 1 followed by manual trip of reactor 2 (other problems), the cause - inter-phase fault on generator transformer	GCR	PWR(34)	
A31	7075	BREACH OF MAINTENANCE SCHEDULE WHEN RETURNING GAS TURBINE 1 TO SERVICE	(GCR) Manual reactor shutdown. Gas turbine returned to service too early before other jobs were finished. The turbine needed to be returned to maintenance	GCR	PWR(40)	
A32	7083	REACTOR AUTOMATIC SHUTDOWN DUE TO LOW REACTOR WATER LEVEL	(BWR), failure of o-ring in the speed control system of the reactor feedwater pump led to low reactor level and automatic trip	BWR	BWR(23)	
A33	7090	DETECTION OF A CONTROL VALVE DELAYED CLOSURE DURING REVIEW OF RECORDED DATA AFTER AN AUTOMATIC REACTOR SHUT DOWN	(PWR) Fault (incorrectly assembled solenoid during backfitting) in FW control valve led to reactor trip on low SG level	PWR	PWR(15)	
A34	7091	REACTOR AUTOMATIC SHUTDOWN DUE TO DEENERGIZED CONTROL POWER FOR INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION	(PWR) control power circuit de-energized due to failed fuses, false signal intermediate range neutron flux high, reactor trip	PWR	PWR(39)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A35	7112	REACTOR CORE ISOLATION COOLING SYSTEM RELIABILITY, 1987-1993 (AEOD/S97-02)	(BWR) generic revision of reliability data	BWR	-	
A36	7126	EXCESSIVE COOLDOWN AND DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM FOLLOWING LOSS OF OFFSITE POWER (NRC Information Notice 95-04, Supplement 1)	(PWR) short to ground of 2 buses, turbine trip, reactor trip, loss of off-site power; additional problem was one of the DGs in maintenance, After the reactor trip, the reactor coolant system rapidly cooled down and depressurized because of a reduction in energy input and an increase in energy removal by full, unthrottled auxiliary feedwater (AFW) flow and several steam flow paths. The steam flow paths included the AFW pump turbine and steam line drains. Because of the loss of offsite power, charging flow and seal injection flow discharge valves failed open, increasing the mass addition to the reactor coolant system. With normal letdown isolated on the safety injection signal, the reactor became water solid. Reactor coolant system pressure increased to the lift setpoint of one of the three pressurizer power-operated relief valves. Over a period of about 6 hours and 15 minutes, the relief valves opened approximately 74 times to control pressure. The pressurizer relief tank rupture disc relieved and reactor coolant system water discharged to the containment. As a common cause, a degraded resistor bushings in the presence of moisture and other contaminants	PWR	PWR(33)	YES
A37	7131	SHUTDOWN SYSTEM NO. 2 (SDS2) TRIP FOLLOWING LIGHTNING STRIKES	(PHWR) Two separate lightning strikes affected two separate transmission lines, full generator load rejection, TG trip and brief interruption of power on 13.8 kV bus, trip of MCP and reactor trip. The load rejection occurred because the CRO did not have enough time to manually reclose unit breaker P3-5 before the second lightning strike occurred. The second strike opened the other unit breaker (P3-4) which effectively disconnected the generator from the NB Power grid to cause the load rejection.	PHWR	PWR(34)	
A38	7136	REACTOR SCRAM DUE TO AN INVASION OF SHRIMP	(PWR) Two circulating water pumps stopped due to high pressure difference signal. Due to the stopping of these pumps, the condenser pressure increased rapidly. Soon after, two main feed water pumps also stopped from the signal of high exhaust pressure from the discharge lines of the turbines of the main feed water pumps. Then the reactor scrammed.	PWR	PWR(30)	
A39	7137	REACTOR SCRAM CAUSED BY THE OPENING OF THE MAIN REACTOR TRIP BREAKER DURING THE PERIODIC TEST	(PWR) Block relay malfunctioned in the reactor protection logic circuit and HE during the test led to spurious reactor trip.	PWR	PWR(39)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A40	7138	REACTOR TRIP DUE TO SOFTWARE OPERATING SYSTEM PROBLEM	(BWR) Software problem led to malfunction of FW control valves, pressere/flow transient and reactor trip. Affected systems: pressure control and FW isolation	BWR	BWR(35)	
A41	7139	REACTOR TRIP DUE TO LOSS OF NON-SAFETY CLASS POWER SUPPLY	Loss of power to non-safety bus led to trip of RCS pump, condensate pump, 2 CCW pumps. Reactor tripped due to low SG level.		PWR(37)	
A42	7140	Turbine trip and reactor scram due to the failure of the main feedwater control valve	(PWR) Failure of the main FW control valve, low SG level led to automatic scram. AFW pump started automatically as required.	PWR	PWR(15)	
A43	7147	ACTUATION OF SECONDARY SHUTDOWN SYSTEM ON FAILURE OF MORE THAN ONE ROD OF PRIMARY SHUTDOWN SYSTEM	(PHWR) Inadvertent actuation of secondary shutdown system (SSS) and reactor trip due to a failure of the solenoid valve, event occurred during the scheduled test of SSS. Additional failure was observed in the primary SS (drop of 2 rods too slow; root cause - inadequate maintenance)	PHWR	PWR(39)	
A44	7158	LEAKAGE FROM TURBINE BUILDING VENTILATION UNIT CAUSES UNIT WIDE LOSS OF ELECTRICAL POWER	(PHWR) Water dripping from a ventilation unit caused an electrical fault and loss of bus that led to a loss of non-essential power at the affected unit (No. 4) and the twin unit (No 2), and subsequent reactor trip. There were several independent failures/faults that contributed to the event (failed unit breaker racking mechanism, leaking valve in condensate system, blocked drain header for the drip trays under the 2 heating and ventilation units, inappropriate caulking of bus duct)	PHWR	PWR(37)	YES
A45	7163	REACTOR SCRAM DUE TO TYPHOON ATTACK	(PWR) Typhoon caused trip of the circuit breaker, loss of power line and power at 2 buses. Due to deficiency in the design the air compressor stopped for 25 min resulting in the of closure of FW valves; trip on low SG level. Accompanied failures in the service water, compressed air, FW control systems	PWR	PWR(37)	YES
A46	7165	REACTOR TRIP DUE TO FAULTY RELAY IN THE PROTECTION SYSTEM OF GENERATOR AND IMPROPER ACTUATION OF CONTROL EQUIPMENT OF THE STEAM GENERATORS	(PWR) Fluctuation in the grid and independent failures in TG protection system (faulty relay) and SG protection led to TG trip and reactor trip on low SG level.	PWR	PWR(34)	
A47	7168	RUPTURE IN EXTRACTION STEAM PIPING AS A RESULT OF FLOW-ACCELERATED CORROSION (NRC Information Notice 97-84)	(PWR) Break of steam line in the 4-th stage regeneration heater; manual reactor scram by operator, plant system responded as expected. However the steam line rupture damaged a non-safety electrical load center and in several cable trays and pipe hangers; root cause was flow-accelerated corrosion in the pipe elbow at a rate higher than expected (limitation of computer calculations)	PWR	PWR(28)	YES
A48	7177	OFFSITE POWER RELIABILITY CHALLENGES FROM INDUSTRY DEREGULATION (NRC Information Notice 98-07)	Generic Rpt/ Decreasing reliability of off-site power sources due to ‘non-utility generation’ and ‘deregulation’. Several events from US industry indicates problems with loss of off-site power		PWR(35) BWR(31)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A49	7180	NUCLEAR POWER PLANT COLD WEATHER PROBLEMS AND PROTECTIVE MEASURES (NRC Information Notice 98-02)	Generic NRC Rpt/ on potential CCF mechanisms of safety related systems (4 events of similar type as 7050); (1) Wolf Creek - Valves plugged on the CCW warming line; (2) – Millstone 2 ice plug in a horizontal common strainer backwash drain line in SW system (both trains inoperable, not declared as LCO); (3) La Salle 2; Ice severed a cable in the transformer cooling logic module that led to the loss of cooling to the main transformer; (4) Mc Guire 2;frozen impulse lines (setpoints for the strip heaters too low) that affected RWST level indications and ability to automatically switch to the ECCS sump/recirculation mode. The NRC report noted 37 cold weather related events at 23 different sites between 1991 and 1997. The study also reported an increasing trend in the number of these events.			YES
A50	7185	REACTOR SCRAM BY AZ-5 PROTECTION ACTION DUE TO AVAILABILITY LOSS OF ECC'S AIR AND HYDRO VOLUMES SUBSYSTEM	LWGR (RBMK) Rupture of air line to safety valve in ECCS (weld defect). Reactor scrammed by operator (AZ-5);	LWGR	PWR(40)	
A51	7196	FORCED OUTAGE DUE TO COMPUTER MALFUNCTION	(PHWR) The plant was tripped automatically by passive control circuitry and no failure of any safety system was involved. Root cause fault of the multiple output connector in the regulating computer	PHWR	PWR(39)	
A52	7197	FORCED OUTAGE DUE TO BOILER FEED WATER RELAY BURNING	(PHWR) Boiler FW pump trip due to relay failure on 'boiler level high' signal, standby pump did not start; load reduction initiated to shutdown the plant but reactor tripped automatically on boiler 'level low' signal	PHWR	PWR(15)	
A53	7198	FORCED OUTAGE DUE TO MODERATOR PUMP TRIP	(PHWR) Event happened due to breakage of one of the lugs connected to moderator pump motor (disconnection of one lead of power supply) and pump trip on over-current protection signal. Following the pump trip the booster rods cooling flow decreased and caused the reactor trip.	PHWR	PWR(39)	
A54	7199	LOSS OF AUXILIARY POWER AT TWO UNITS OF BALAKOVO DUE TO BREAKER FAILURE IN THE 220KV OPEN SWITCHGEAR	(WWER-1000) A single phase short circuit in the high voltage breaker (HVB) of the transformer of Unit 1 that led to unit trip, and subsequently to trip of Unit 3. When restoring the power to the plant the HVB developed single-phase short-circuit again which turn to 2-phase short-circuit and asymmetrical voltage to auxiliary power loads (causing damage to four 6kV motors and eleven 0.4kV motors). Power was supplied from Unit 2 for ~4 hrs before restoration of power to both units through auxiliary unit transformers.	WWER	PWR(35)	YES

#	**IRS Number**	**Title**	**Description**	**NPP type**	**IE type (EPRI)**	**In-depth analysis ?**
A55	7200	UNIT SCRAM ON TRIPPING OF TWO TURBINE GENERATORS DUE TO PERSONNEL WRONG ACTIONS	LWGR (RBMK) HE during test of insulation performance of the arrester of the standby transformer (work under voltage) led to a single short-circuit; bus duct protection was not successful and transformer protection worked leading to the disconnection of two TGs from the grid. DGs started (1/4 failed to start); Independent multiple failure or errors	LWGR	PWR(34)	
A56	7209	INADVERTENT REACTOR BUILDING CONTAINMENT SPRAYING	(PWR) Surveillance test of the Containment Spray system and Component Cooling system MOVs, start-up of CS train signal and containment isolation demand; Additional failures/malfunctions were CR stack in upper position, and re-setting of cont. isol. signal ineffective. RCP shutdown due to loss of CC water supply.	PWR	PWR(31)	
A57	7225	TURBINE TRIP DURING A CONTROL VALVE MOVABILITY TEST SUBSEQUENTLY LEADING TO A REACTOR SCRAM	WWER-440 test of TB control valves; TB trip on a low oil pressure signal. The second TB triped due to SG level too high (due to leaking FW control valves), Reactor trip on last TB trip.	WWER	PWR(33)	
A58	7227	REACTOR PROTECTION ACTUATION DUE TO I&C TECHNICIAN'S ERRONEOUS MANIPULATION AND SUBSEQUENT MANUAL REACTOR SCRAM	(WWER-440) HE during the startup tests (of the Automatic Power Controller) generated spurious RPS signal AZ-3 followed by operator's manual scram	WWER	PWR(40)	
A59	7236	MANUAL REACTOR SCRAM DUE TO FAILURE OF SAFETY SYSTEM TRAIN 2 CAUSED BY ELECTRICAL EQUIPMENT FLOODING BY THE FIRE EXTINGUISHING SYSTEM WATER	(WWER-1000) Spurious signal of fire distinguishing system, malfunction of electrical equipment of one safety system train; operator trips the reactor direct cause: short circuit; root causes: poor design of the DIP-1 fire detectors and the poor quality of the components of the logic control unit the technical characteristics of which are not suitable for the working environment ; RPS and ESFAS affected (degraded also due to loss of power)	WWER	PWR(40)	YES
A60	7241	REACTOR TRIP AND SUBSEQUENT POST-TRIP COOLING DEFICIENCIES	(GCR) Reactor tripped following a feed valve trip on one of the boilers (SGs) after a control relay coil failure (operating at the extreme limits of its voltage tolerance). There were several independent failures in the post trip cooling system: standby pump did not start (HE- wrong selection switch on the reactor desk), no indication of emergency feeding pump operation. In consequence the post trip capabilities were degraded.	GCR	PWR(15)	YES
A61	7242	OPEN SAFETY RELIEF VALVE WITH REACTOR AT POWER OPERATION	(BWR) test on the main steam radiation monitoring system; Operator erroneously opened a safety relief valve (SRV) instead of the drain-valve. Restoration of this error to prevent scram was not successful (too long) and reactor trip occurred on signal "suppression pool temperature high".	BWR	BWR(11)	
A62	7244	SMOKE NEAR HIGH PRESSURE TURBINE EXHAUST PIPE AND MINOR FLAME NEAR GENERATOR SLIP RING	(PHWR) Oil leak in the TB caused smoke (HEs), TB and reactor were tripped manually	PHWR	PWR(33)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A63	7247	SMALL LOSS OF PRIMARY COOLANT ACCIDENT THROUGH THE FAILED CHECK VALVE OF PRIMARY CIRCUIT PRESSURE CONTROL SYSTEM.	(WWER -1000) Failed check valve in the PRZ system (YP13S01) led to small LOCA to the containment	WWER	PWR(5)	
A64	7264	LOSS OF CONSENSER VACUUM FOR JOINT CRACK PRODUCES TURBINE TRIP, REACTOR SCRAM, SIGNAL TO CLOSE MAIN STEAM ISOLATION VALVES FOR LOW CONDENSER VACUUM FAILS FOR WATER CONDENSATION ON CONDENSER PRESSURE SENSING LINES	(BWR) Loss of condenser vacuum due to a crack in the joint. Reducing reactor power did not prevent trip as the condenser pressure increased significantly. Condenser bypass was automatically closed (as required). However, MSIV failed to close automatically due to condensation plugs in the sensing lines (effect of the transient and design deficiency). It was closed manually by operator. In consequence the post trip mitigation capabilities were degraded. This is a potential source for failure of steam pressure sensing lines: condensation plugging. This is a risk well known, design standards contain precautions to keep certain slope to allow water draining, to avoid too long horizontal segments, etc. What has been learnt in this event is that sensing lines working properly at stable conditions, even at mild transients, may experience a faulty performance at strong transients because a fast pressure increase can produce water condensation at a rate higher than the tubing draining capability.	BWR	PWR(27)	YES
A65	7265	SPURIOUS OPENING OF PRESSURISER POWER OPERATED RELIEF VALVES RESULTING IN REACTOR SCRAM	(PWR) 2/3 PORVs began spuriously open and close. Operator closed isolation valves but it was in conflict with TS. Exchange was made of one electronic circuitry but it did not help (breach of TS requirements and inadequate following the procedures). After repair both PORVs opened spuriously again, and the reactor tripped on low pressure. Direct cause was failed power supply module in the valve actuation circuitry (on the personalization card). Dependent multiple failures and errors	PWR	PWR(7)	YES
A66	7267	LOSS OF LEAK-TIGHTNESS OF REFUELING MACHINE COUPLING MODULE DURING REFUELING	(LWGR/RBMK). Leakage from the refueling machine coupling module during on-line refueling (piping weld fault). Reactor tripped manually AZ-1.	LWGR	PWR(5)	
A67	7274	INADEQUATE OR POORLY CONTROLLED, NON-SAFETY-RELATED MAINTENANCE ACTIVITIES UNNECESSARILY CHALLENGED SAFETY SYSTEMS (NRC INFORMATION NOTICE 98-36)	Generic NRC report regarding poorly controlled maintenance that unnecessarily challenges the safety systems. Several cases of electrical problems (e.g. loss of station or auxiliary unit transformers, off-site power, reactor trip) due to phase-to-ground arc. These included roof repair material left on the roof, unattended water leakage, gasket material missing, condensation in the duct, long term degradation of insulation, etc.)			YES*
A68	7284	REACTOR TRIP DURING THE REPLACEMENT OF TRANSMISSION LINE PROTECTION PANEL	(PWR) Work on the replacement of a transmission line protection panel. Generator and main transformer tripping relay were actuated causing the trip of both the operating generator and TB and automatic reactor scram. This was due to error in connecting two wires in the breaker protection module	PWR	PWR(34)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A69	7286	REACTOR TRIP DUE TO HEAT TRANSPORT HIGH PRESSURE DURING HEAT TRANSPORT INTERCONNECTION VALVE TEST	(PHWR) During a routine test of HT system loop interconnection valve a pressure peak occurred and resulted in reactor trip (the cause - incorrect test procedure)	PHWR	PWR(8)	
A70	7292	DECLARATION OF SITE EMERGENCY AT HUNTERSTON B FOLLOWING TWO COMPLETE LOSSES OF ELECTRICAL GRID SUPPLY DURING A PERIOD OF BAD WEATHER.	(GCR) High wind conditions led to loss of off-site power in 2 reactors. Post trip cooling established successfully and grid restored in about 1 hr , normal cooling reinstated and diesel generators placed back onto standby (in 2 hrs from IE). Several hrs later there was another loss of off-site power and in this time manual reconfiguration of essential electrical supplies was needed to be done manually because the automatic system had not been fully reset after the first initiator; as a result various cooling services were not immediately available to the reactors. It was the reduction in the functionality of the normal cooling supply equipment	GCR	PWR(35)	YES
A71	7301	REACTOR SCRAM CAUSED BY SECONDARY TRANSIENT AND FOLLOWED BY CONTROL ROD SEIZURE	(WWER-440) Unit was at low power to perform corrective maintenance on the transformer following a ground fault. Instability of FW supply to SGs led to plant trip. Instability was caused by leaky valve in the control of FW make-up flow to FW tank and the use of low pressure pre-heater line for this make-up with a gate valve which appeared to be eroded. Trip of the plant was manual (because RPS was disabled to allow for maintenance of the generator). Additional failure was with CR which stack in high position. Potential impact through maintenance induced faults	WWER	PWR(16)	
A72	7312	AUTOMATIC SHUTDOWN OF UNIT 6 OF THE KASHIWAZAKI-KARIWA NPS	(BWR) Unit trip automatically after generator trip due to actuation of 500 kV line protection relay. Root cause was the reverse polarity of current transformers of the concerned relay and lightning strike	BWR	PWR(34)	
A73	7313	AUTOMATIC SHUTDOWN OF THE NO. 6 UNIT OF KASHIWAZAKI KARIWA NPP	(BWR) the generator and TB tripped on the signal “generator excitation system shutdown”. This was a deficiency in the excitation system monitoring programme.	BWR	PWR(34)	
A74	7314	AUTOMATIC SHUTDOWN OF UNIT-3 OF THE FUKUSHIMA DAIICHI NPS	(BWR) “Neutron flux high” false signal generated due to a lighting strike through the vent stack duct adjacent to APRM transmission cables (installed in the same cable route)	BWR	BWR(35)	
A75	7323	RUPTURE OF THE SHELL SIDE OF A FEEDWATER HEATER AT THE POINT BEACH NUCLEAR PLANT (NRC INFORMATION NOTICE 99-19)	(Peach Bottom) Manual reactor trip because of the steam leak that resulted from the failed heater shell. The licensee did not have in place an inspection program for examining the thickness of the walls of feedwater heaters. Similar failures of feedwater heaters had previously occurred at the Dresden Nuclear Power Station in 1983 and as recently as January 18, 1999, at the Susquehanna Steam Electric Station and April 28, 1999, at the Pilgrim Station		PWR(40) ?	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A76	7327	TOTAL LOSS OF ESSENTIAL AND AUXILIARY SERVICE WATER SYSTEMS	(BWR) Loss of service water flow (for about 6 minutes) due to a HE (inadvertent closure of 2 gate valve at water intake when conducting periodical quarterly test of inlet gate valves) followed by reactor trip due to increased temp of oil in the frequency inverters for 2 recirculation pumps	BWR	BWR(22)	
A77	7330	INSTABILITY EVENT IN OSKARSHAMN NPP UNIT 3	(BWR) Reactor trip on high power signal caused by power oscilation (unfavorable axial power distribution, lack of monitoring equipment to detect such unstable conditions)	BWR	BWR(28)	
A78	7333	SUDDEN DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM AND ACTUATION OF SAFETY INJECTION SYSTEM	(PWR) Reactor trip on RCS pressure low and ECCS injection followed by opening of PORV. Cause was a malfunction of PRZ pressure control system (deficiency not detected during the tests due to insufficient comprehensiveness of the test)	PWR	PWR(6)	
A79	7334	AUGMENTED INSPECTION TEAM - REACTOR TRIP WITH COMPLICATIONS AT INDIAN POINT 2 (INSPECTION REPORT NO. 50-247/99-08)	(PWR) Reactor trip on “over-temperature delta temperature” signal and 3 minutes later the normal off-site power to all 4 480V vital buses was lost due to breakers trip. Additionally one of the DG breakers triped leaving vital bus 6-A de-energized, loss of power to AFW pump, battery charger and some emergency core cooling components. Battery was discharged in 7 hrs leading to a loss of control DC power to safety related systems including AFW pumps. This was the result of several independent failures (or deficiencies in setpoints)	PWR	PWR(37)	YES
A80	7342-1	PARTIAL LOSS OF SAFEGUARD SYSTEMS AS A RESULT OF EXTERNAL FLOODING	See below		-	
A81	7342-2	PARTIAL LOSS OF SAFEGUARD SYSTEMS AS A RESULT OF EXTERNAL FLOODING	(PWR) Weather condition caused external flooding of many rooms of the NPP nuclear island. Additionaly off-site grid was lost for several hours. Degradation of systems required to assure primary coolant inventory and core cooling due to flood and degradation of essential support systems. Common cause failure (including potential for CCF).	PWR	PWR(34)	YES
A82	7348	REACTOR SHUTDOWN DUE TO "TURBINE SOLENOID VALVE ACTUATION" SIGNAL	(PWR) Automatic reactor trip by automatic turbine trip signal due to emergency trip oil pressure dropped caused by an oil leak into the recovery drain (failure of o-ring seal)	PWR	PWR(33)	
A83	7355	ENTRY OF FOREIGN MATERIAL IN THE PRIMARY HEAT TRANSPORT SYSTEM IN PRESSURISED HEAVY WATER REACTORS (PHWRS) IN INDIA	(PHWR) Manual scram due to entry of foreign material into the primary HT system.	PHWR	PWR(40)	
A84	7362	OPERATIONAL ISSUES IDENTIFIED IN BOILING WATER REACTOR TRIP AND TRANSIENT (NRC INFORMATION NOTICE 2000-01)	(BWR) Automatic reactor scram on low reactor level signal after a partial loss of feedwater. Inadvertent closure of FW valve due to a problem with valve control switch leading to initiation of safeguards, high level in the reactor and postpone trip of high pressure core injection (HPCI). Several operational performance issues complicated the transient and recovery.	BWR	BWR(24)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A85	7365	LOSS OF POWER OF A 220 VAC SAFETY BUS BAR	(PHWR) Manual trip due to electrical system failure (inverter failure probably due to a printed circuit board mis-function or mis-calibration and an outlet filter capacity lower than original value. Transient caused a turbine trip because of speed turbine measure loss and the condenser steam discharge valves blockage (fail to open causing the steam by-pass loss) due to electrical supply loss to Electric Hydraulic Control system, automatic power reduction, demand for opening of the four atmosphere steam discharge valves (ASDV) and eight main steam safety valves (MSSV) actuation. The reactor was tripped manually when power reached 12 % FP. Loss of vital instrumentation AC and control AC, communication and alarm annunciation, and turbine by-pass.	PHWR	PWR(37)	YES
A86	7371	ACTUATION OF AZ-1 EMERGENCY PROTECTION DUE TO REACTOR OPERATOR ERROR WHILE PERFORMING SWITCH-OVERS	(PWR) Automatic reactor trip on “last turbine off” due to operator error when performing RPS test (active channel received simulated source range trip signal)	PWR	PWR(34)	
A87	7377	LOSS OF AUXILIARY POWER IN UNITS 3 AND 4 OF LENINGRAD NPP AS A RESULT OF EQUIPMENT DAMAGE IN THE OPEN SWITCHGEAR CAUSED BY STORMY WINDS	(WWER) Automatic reactor caused by TG trip due to generator protection system. Cause was high wind (and the fall of a lightning arrester due to wind). Power was also lost in other twin unit (in cold shutdown) due to electrical failures (in aux transformer system) caused by wind.	WWER	PWR(34)	
A88	7384	FEED WATER TRANSIENT IN OSKARSHAMN-2 NPP UNIT	(BWR) External voltage drop in the grid led to a disconnection of TG from the grid and heavy power/pressure transient, partial scram by operator followed by automatic scram	BWR	BWR(1)	
A89	7388	STEAM GENERATOR TUBE FAILURE AT INDIAN POINT UNIT 2 (NRC INFORMATION NOTICE 2000-09)	(PWR) SG tube rupture and manual reactor trip	PWR	PWR(26)	
A90	7395-1	BELOYARSK NPP UNIT 3 SCRAM DUE TO EVENTS IN THE OPERATION OF THE URALS POWER SYSTEM	(FBR) Disturbances in the off-site grid (voltage and frequency oscillations). Rotation speed of MCPs led to automatic reactor trip	FBR	PWR(33)	
A91	7395-2	TOTAL LOSS OF OFF-SITE POWER AT THE BELOYARSKY NUCLEAR POWER PLANT OWING TO A POWER SYSTEM OPERATING PROBLEM	As above			
A92	7400	UNIT DISCONNECTED FROM THE GRID AND REACTOR SHUT DOWN BY THE EMERGENCY PROTECTION SYSTEM AS A RESULT OF A LIGHTNING STRIKE IN A 24 KV CONDUCTOR	(WWER-1000) Lightning strike caused a short circuit in 24 kV conductor and disconnection of unit from the grid. TG trip followed. There were additional failures or design/operational weaknesses (circuit breakers, pressure measurement in the make-up system) that caused intiation of ECCS by operators	WWER	PWR(39)	
A93	7413	REACTOR SCRAM RESULTING FROM MAIN TRANSFORMER PROTECTION	(PWR) Main transformer protection tripped reactor and turbine. Direct cause was fuse burned due to overheating (root cause – weakness in the testing/maintenance of the main transformer)	PWR	PWR(39)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A94	7415	FORCED SHUTDOWN DUE TO INCREASED HEAVY WATER LOSS FROM PRIMARY HEAT TRANSPORT SYSTEM	(PHWR) Manual trip to identify D2O leakage	PHWR	PWR(40)	
A95	7416	FORCED SHUTDOWN DUE TO COMPUTER MALFUNCTION	(PHWR) Reactor automatic trip on false 'high temperature' in one channel signal (due to a faulty filter capacitor in the regulating channel A)	PHWR	PWR(39)	
A96	7418	MODERATOR WATER LEAK FROM SUCTION FLANGE OF MODERATOR PUMP AT NAPS-2	(PHWR) Manual reactor trip due to heavy water leakage from one of the moderator pumps (direct cause – a missing stud and 2 loose studs in the pump flange joint)	PHWR	PWR(40)	
A97	7419	INGRESS OF PRIMARY HEAT TRANSPORT HEAVY WATER (D2O) TO EMERGENCY CORE COOLING SYSTEM HEAVY WATER ACCUMULATOR DURING EMERGENCY CORE COOLING SYSTEM MONTHLY TEST	(PHWR) Leakage of primary coolant into ECCS system on "low PHT pressure" signal. Direct cause - a fault in functionality of two valves (drifting of limit switch setting and check valve failure to re-seat)	PHWR	PWR(39)	
A98	7423	AUTOMATIC SHUTDOWN OF MIHAMA POWER STATION UNIT-2	(PWR) Automatic trip of the generator and reactor trip by loss of generator field. Direct cause – faulty contact of one of the cable terminals	PWR	PWR(34)	
A99	7424	MANUAL SHUTDOWN OF MIHAMA POWER STATION UNIT-2	(PWR) Manual reactor shutdown to resolve leakage problem in CVCS letdown piping (fatigue fracture in the weld due to cavitation and vibration)	PWR	PWR(40)	
A100	7425	MANUAL SHUTDOWN OF OHI POWER STATION NO. 2	(PWR) Manual shutdown of turbine followed by reactor trip due to problem with decreasing condenser vacuum (sea water leakage into the condenser). This was incorrect operator response because he made error in reading the vacuum indication (turbine output instead of condenser vaccum reading)	PWR	PWR(40)	
A101	7426	MANUAL SHUTDOWN IN RESPONSE TO A RISE IN THE IODINE CONCENTRATION IN THE REACTOR COOLANT	(BWR) Manual shutdown in response to radioactivity increase in the coolant due to fuel element failures (2 elements replaced)	BWR	BWR(36)	
A102	7427	REACTOR TRIP DUE TO STEAM GENERATOR C LEVEL DURING START-UP	(PWR) During start-up of the unit one of the turbine bypass valves (Group I) stack open in addition to opened 6 valves of group II. This caused excessive steam release and SG pressure drop and SG level low that followed by turbine trip and reactor trip	PWR	PWR(29)	
A103	7429	REACTOR TRIP BY THE STEAM GENERATOR LOW LEVEL SIGNAL	(PHWR) During the periodical test of SG protection channel a reactor trip signal was generated on low SG level. The direct cause was the drain valves internal leakage in the transmitter test loop in the remaining SGs	PHWR	PWR(39)	
A104	7431	REACTOR TRIP BY DEAERATOR STORAGE TANK LEVEL CONTROLLER FAILURE	(PWR) Reactor automatic trip on SG low level signal. Direct cause was a false signal "deaerator storage tank level low" that led to trip of booster and the main FW pumps and low SG level	PWR	PWR(39)	
A105	7437	MANUAL SHUTDOWN OF IKATA POWER STATION UNIT-1	(PWR) Manual shutdown to inspect/resolve a leakage problem in the body of a valve in the secondary cooling system (chloride stress corrosion crack)	PWR	PWR(40)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A106	7446	MANUAL SHUTDOWN OF FUKUSHIMA DAIICHI NUCLEAR POWER STATION UNIT 6	(BWR) Manual shutdown to inspect/resolve problem of increased flow rate in the gaseous waste treatment system	BWR	BWR(36)	
A107	7447	MANUAL SHUTDOWN OF TOKAI DAINI POWER STATION	(BWR) Manual shutdown to inspect/resolve problem of increased flow rate in the pump seals (2 primary loop recirculation pumps)	BWR	BWR(36)	
A108	7448	MANUAL SHUTDOWN OF THE FUKUSHIMA DAINI NUCLEAR POWER STATION UNIT 4	(BWR) Increased radiation level in the reactor coolant system due to leaking fuel element led to manual shutdown to resolve the problem	BWR	BWR(36)	
A109	7456	REACTOR SCRAM BY AZ-1 EMERGENCY PROTECTION DURING UNIT SHUTDOWN DUE TO PERSONNEL ERRORS	(WWER-440) During unscheduled shutdowm to inspect SG leakage automatic trip occurred due to operator error	WWER	PWR(39)	
A110	7461	DISCONNECTION OF UNITS 2 AND 3 FROM THE GRID AND TRIGGERING OF EMERGENCY PROTECTION SYSTEM	(PWR) ? When dismantling a crane in the construction site of Unit 4 damage was made to 6 kV cables and transformers. Other units at the site were disconnected from the grid to localize the short circuit	PWR	PWR(39)	
A111	7463	UNIT SHUTDOWN DUE TO PERSONNEL ERRORS	LWGR (RBMK) Turbine generator trip initiated erroneously by operator (initiation of oil fire algorithm by resetting an untagged key to activated position)	LWGR	BWR(34)	
A112	7473	MANUAL SHUTDOWN FOLLOWING LEAKAGE OF REACTOR BUILDING CLOSED COOLING WATER INSIDE THE REACTOR CONTAINMENT VESSEL	(BWR) Manual shutdown to deal with a leakage in the containment closed cooling water system (leakage through a valve gland)	BWR	BWR(36)	
A113	7474	ANGRA 1 MANUAL SHUTDOWN DUE TO NITROGEN LOW PRESSURE IN MAIN STEAM ISOLATION VALVE ACTUATOR	(PWR) Manual shutdown of the unit to replace o-ring in the nitrogen pressure tank of the MSIV actuator (additional error in the interpretation of TS requirements)	PWR	PWR(40)	
A114	7475	MANUAL SHUTDOWN FOLLOWING LEAKAGE OF TURBINE VALVE CONTROL OIL	(BWR) Manual shutdown of the unit to resolve an oil leakage problem in the turbine control oil system	BWR	BWR(36)	
A115	7482	MANUAL SHUTDOWN DUE TO THE FAILURE OF THE MECHANICAL SEAL FOR THE PRIMARY LOOP RECIRCULATION PUMP	(BWR) Manual shutdown of the unit to resolve a failure of the mechanical seal in the primary loop recirculation pump	BWR	BWR(36)	
A116	7487	REACTOR SHUT DOWN BY FAST-ACTING EMERGENCY PROTECTION SYSTEM OWING TO A SPURIOUS SIGNAL FOR A PRESSURE INCREASE IN THE LOWER WATER LINE ROOMS	LWGR(RBMK) Reactor automatic trip on false signal “pressure increase in in the lower water line room” generated by commissioning work on the cable penetration	LWGR	BWR(35)	
A117	7488	DROPPED CONTROL ROD AND LOSS OF OFFSITE POWER	(PWR) Periodic test of CR operation a CR dropped int the core causing scram accompanied by a total loss of electricity supply to the grid. On-site power was from DGs, SG safety valves opened on demand after 45 min whereupon safety injection initiated automatically	PWR	PWR(35)	
A118	7490	REACTOR SCRAM DUE TO LOSS OF 48V DC POWER SUPPLY	(PWR) Reactor scram during periodic test and maintenance of DC power system rectifier due to a HE (wrong switch opened causing bus de-energized)	PWR	PWR(39)	

#	**IRS Number**	**Title**	**Description**	**NPP type**	**IE type (EPRI)**	**In-depth analysis ?**
A119	7491	DUE TO POWER LOSS OF PROTECTION SYSTEM RESULTING IN REACTOR COOLANT PUMP A/B TRIP SIGNAL ACTUATION, CAUSE REACTOR AND TURBINE TRIPPED AUTOMATICALLY	(PWR) During I&C visual inspection of a protection cabinet a input cable to one of the breakers lost contact due to an inadvertent human action (cable termination faulty) that led to reactor trip	PWR	PWR(39)	
A120	7492	INCIDENT OF LIGHT WATER LEAKAGE FROM END-SHIELD COOLING SYSTEM IN KAPS-2.	(PHWR) Manual reactor shutdown to resolve problem of leakage of light water from the end-shield cooling system (front compartment). Leakage was due to improper isolation of check valve and led to increase gamma radiation level	PHWR	PWR(40)	
A121	7493	REACTOR TRIP ON 'PRIMARY HEAT TRANSPORT PRESSURE HIGH' DUE TO MALFUNCTION OF PHT PRESSURE CONTROL SYSTEM AT KAPS-2	(PHWR) Automatic reactor trip on ‘high PHT pressure’ occurred during testing of gasket that had been replaced. It was due to malfunction of a relay in the pressure control system and a broken wire.	PHWR	PWR(39)	
A122	7499	DISRUPTIONS IN NUCLEAR POWER PLANT OPERATION CAUSED BY FOREIGN OBJECTS IN NUCLEAR POWER PLANT SYSTEM PIPING AND EQUIPMENT	(RBMK and WWERs) Generic report – reference to events at 7 plants. In one plant there was an event during scheduled maintenance with the reactor subcritical (out of normal flow rate range in some fuel channels). It was flow blockage by a piece of white cloth (cleaning material). In other plants there were metal objects and polythene bag in the inlet reactor header and fragments of a rubber seal. There were foreign objects also in other systems including oil.	LWGR		YES*
A123	7500	UNIT SCRAM THROUGH ACTUATION OF EMERGENCY PROTECTION SYSTEM 1 DURING A CHECK ON THE REACTOR COOLANT PUMP PRESSURE SENSOR	(WWERE-440) While the operation of the measurement channel of the reactor's emergency protection system was being restored (problem with RCP pressure difference signal), a spurious signal was generated by the reactor's emergency protection system due to concealed damage to the control cable insulation.	WWER	PWR(39)	
A124	7504	REACTOR SCRAM BY AZ-1 PROTECTION ON THE SIGNAL OF "SIMULTANEOUS LOSS OF 380 V VOLTAGE TO RPS DISTRIBUTION IN TWO LEADS" WHILE THE VOLTAGE WAS ACTUALLY AVAILABLE	(WWER-440) Automatic reactor trip due to a false signal of loss of voltage in RPS. Problem caused by a design deficiency made during modification	WWER	PWR(39)	
A125	7511	TURBINE TRIP FOLLOWED BY REACTOR SHUTDOWN AND SAFETY INJECTION DUE TO STEAM GENERATORS OVERFILL	(PWR) Maintenance on one of the electrical equipment cabinets a transfer to house-load operation was initiated (due to turbine overspeed protection system) and due to a delay in the closure of main turbine valve turbine was tripped followed by reactor trip.	PWR	PWR(33)	
A126	7517	DIABLO CANYON MANUAL REACTOR TRIP AND STEAM GENERATOR WATER LEVEL SETPOINT UNCERTAINTIES (NRC INFORMATION NOTICE 2002-10, SUPPLEMENT 1)	(PWR) manual reactor trip which resulted from a failure of the main feedwater regulating valve, non-conservative steam generator setpoints and contributing causes.	PWR	PWR(15)	
A127	7524	FOREIGN MATERIAL IN STANDBY LIQUID CONTROL STORAGE TANKS (NRC INFORMATION NOTICE 2002-05)	(BWR) Plastic material debris found in the Standby liquid control system storage tank. It was remaining of the wrapper material for the chemicals.	BWR		YES*

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A128	7530	SIMULTANEOUS LOSS OF TRAIN A CONTROBLOC AND TRAIN A ELECTRICAL SWITCHBOARDS FOR THE 6.6 KV AC NORMAL DISTRIBUTION SYSTEM AND 6.6 KV AC EMERGENCY SUPPLIED DISTRIBUTION SYSTEM	(PWR) Re-qualification tests were carried out on the inverters that supplied power to uninterruptible electrical system. Failure during this test led to loss of respective switchboard, loss of one of the 2 control system train (A) and loss of 6 kV switchboard, trip of one of the RCPs and reactor trip. Several safety relevant systems were affected: CVCS: Chemical Volume and Control System for the reactor coolant system. This system ensures also water injection in the reactor coolant pump seals. It comprises two charging pumps (one for each train) and an auxiliary pump, which guarantee the injection flow to the reactor coolant pump seals when both charging pumps are down. AFWS: Auxiliary Feedwater System to steam generators, equipped, for both Train A and Train B, with a pump set energized via the LHA/B switchboard, and a turbine-driven pump powered by steam from the steam generators. RCP seal injection backup: a turbine-driven generator powered by steam from the steam generators, when necessary, the electrical power supply to the CVCS auxiliary pump. CCWS: Component Cooling Water System, an intermediate two trains cooling system that cools notably the thermal barrier for the four reactor coolant pumps. One of two subsystems was failed due to IE. The other was put into operation because of the test (so it was not on standby)	PWR	PWR(37)	YES
A129	7530	SIMULTANEOUS FAILURE OF THE CONTROBLOC, TRAIN A AND 6.6 KV EMERGENCY AND NON-EMERGENCY SWITCHBOARDS, TRAIN A	As above			
A130	7533	MANUAL SCRAM FOLLOWING LOSS OF 400 KV LINE	(WWER-440) A short-time non-symmetry at 400 kV line led to protection programme F46 in the electrical system (disconnection 400 kV line) followed by automatic transfer to back-up power supply. TGs sucessfully rejected load to house-load but the transient caused one MFW pump trip and reduced availability of condensate pumps. Operators manually tripped the reactor because of confusion (unnecessarily). Short-circuit in the 400 kV Lemesany distribution station occurred due to an erroneous switch-on of grounding knife at the V478 line outlet.	WWER	PWR(40)	YES
A131	7536	STEAM CONDENSATION-INDUCED WATER HAMMER IN A DUMP STEAM LINE, LEADING TO FAILURE OF REACTOR COOLING WATER PIPEWORK AND SHUTDOWN OF REACTOR 2	(GCR) operations to transfer the dump steam from Unit 1 reactor (at low power due to a problem with the overspeed gear) to Unit 2 condensers, using the interconnectors between the dump steam systems of the two reactors. Significant or unforeseen interaction between systems (the accumulation of condensate in both the dump steam lines and water hammer). It led to a fracture of a pipe in the reactor cooling water system	GCR		YES

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A132	7538	SHUTDOWN OF UNIT 2 DUE TO FIRE IN THE PRIMARY CIRCUIT CABLE SHAFT A110/2	(WWER-440) Fire in primary cable room in the reactor building A110/2 (with smoke spreading to 3 other rooms), several false signals and RCP trip. Reduction of power and unit shutdown in about 30 minutes. Cables in this room were dedicated mostly to monitoring and measuring functions not safety-related. False signals were probably due to smoke effect and simultaneous short circuit of different cables. The fire led to loss of power of the valves on make-up water header TK52 and on one leg of the reactor water cleanup system as well as to loss of position indication of some of the Y system isolation valves. Several process parameter measuring circuit failed or non-real values were indicated	WWER	PWR(41)	YES
A133	7539	REACTOR TRIP DUE TO FIRE IN TURBINE	(PWR) Oil fire due to a leak from the turbine bearing line. Manual reactor trip.	PWR	PWR(40)	
A134	7543	ENGINEERED SAFETY FEATURES ACTUATION DURING A TRANSIENT INITIATED BY GENERATOR TRIP	(WWER-1000) TG disconnected from the grid due to a spurious signal during the generator protection circuit test. The limiting system reduced the reactor power from 100 to 38% but there were some failures in the system in response to transient: only 2/5 steam dump stations to the condenser were opened, turbine driven MFW pumps failed to transfer from normal steam supply to alternate steam supply, only 1/2 AFW pumps started to supply water to SGs. Reactor tripped due to decrease of water inventory in SGs. EFW actuated automatically but there were additional failures (e.g. SG3 isolated at the feedwater side). The main root causes were ineffective detection and deficient restoration of equipment. Multiple failures. RCP – failure in electronic module; Reduction in feedwater flow from turbine-driven feed pumps - drop in the rotation speed of turbine-driven feed pumps 1 and 2 owing to infiltration of moisture into the casing of the pump turbodrives (deficiencies of the design of the draining system); cut-out of the main feed regulator: mismatch in the sensor signals for the feedwater flow to steam generator 1 - deficiencies in the design, Accident precursor.	WWER	PWR(34)	
A135	7546	UNIT SCRAM BY EMERGENCY PROTECTION SYSTEM 1 OWING TO REACTOR COOLANT PUMP TRIP	(WWER-1000) Automatic reactor scram after trips of RCP2 (power reduction) RCP4 (by protection system on low SG level), and RCP1 (by protection system on SG level high). Causes - failure of electronic module (trip of RCP2), design deficiencies of the drainage system of the auxiliary header (trip of RCP4), deficiencies in the design of feedwater control system (trip of RCP1) (mismatch in the sensor signals for the feedwater flow to SG at certain flow range). Degradation of essential support systems including condensate and feedwater, feedwater control system. Accident precursor.	WWER	PWR(14)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A136	7555	PRESENCE OF FOREIGN MATERIAL IN THE PRIMARY HEAT TRANSPORT SYSTEM OF RAJASTHAN-3	(PHWR) Reactor was tripped manually when flow rate in one of the core channels was detected. The cause was a presence of foreign material (galvanized iron disc of 0.8 mm thickness and about 75 mm diameter which was used as welding dam during construction)	PHWR	PWR(40)	
A137	7561	SAFETY INJECTION ACTUATED BY "VERY LOW PRESSURISER PRESSURE " PROTECTION CAUSED BY INAPPROPRIATE OPERATOR MANOEUVRE	(PWR) Inadvertent actuation of safety injection and reactor trip. Direct cause is that the operator mistakenly dis-inhibited the interlock of the automatic start up of the safety injection system under very low reactor coolant pressure) Transient challenged the reactor and secondary cooling systems beyond the limits authorized by the technical operating requirements: the subcooling margin (DT sat) reached nearly 170°C and the difference between the reactor coolant and secondary pressures exceeded 150 bar (the pressurizer relief valve opened 75 times). Potential LOCA and SGTR due to IE.	PWR	PWR(9)	YES
A138	7562	UNPLANNED CHANGES IN NUCLEAR POWER PLANT UNIT POWER DUE TO PERSONNEL ERRORS	(WWER-1000) Incorrect actions by electrical workshop repair personnel during maintenance of the fire pump (failure to disconnect a relay of the pump from the whole circuit of the 2BA01 bus) causing loss of power to some safety related equipment (the reactor coolant pump; the condensate extraction pump; the coolant circulating pump; and the service water pump)	WWER	PWR(39)	YES
A139	7570	NEUTRON FLUX OSCILLATIONS AFTER BYPASS OF HIGH PRESSURE FEEDWATER HEATERS	(BWR) Automatic reactor scram. Event was during preparation for a mid-cycle plant shutdown to repair two check valves. In compliance with the procedures the moisture separators and reheaters between the high pressure turbine and the low pressure turbine were taken out of service while the plant operated still in full power. One of the two high pressure feedwater heaters trains was bypasses (faulty actuation of the drain tank level limit value "too high" due to erroneous instrument signal). This resulted in asymmetry of water temperature in the core and power rise to 120% followed by scram. Significant or unforeseen interaction between systems.	BWR	BWR(34)	YES
A140	7571	UNAVAILABILITY OF 1 OUT OF 4 MAIN STEAM SAFETY AND RELIEF STATION	(PWR) Manual turbine trip to investigate turbine oil system followed by inadvertent closure of turbine bypass by operator, increase of the steam pressure and automatic reactor trip. Main steam pressure relief system was affected (additional faults). It was found that 1/4 SG main steam SVs and RVs (in SG3) was not operable due to HE (Pilot valve lines left closed during the last shutdown).	PWR	PWR(36) ?	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A141	7575	REACTOR SCRAM AND FAILURE OF THE INJECTION VALVE OF THE REACTOR CORE ISOLATION COOLING SYSTEM AT LAGUNA VERDE NPP-1	(BWR) After maintenance on main steam system (fixing steam leaks and thermal insulation of the bypass valve) operators started to increase power when RPS trip signal was generated in one channel (bus B) (high pressure vessel). It was followed by reactor scram, trip of 2 FW pumps and manual start of the reactor core isolation cooling system but the injection valve closed automatically and could not be opened manually (cause – incorrect wiring in the control circuit). Operator initiated high pressure core spray system. Multiple independent failures (failures in the human performance + hardware fault).	BWR	BWR(14)	
A142	7581	SHUTDOWN OF TWO REACTORS FOLLOWING FAILURE OF GAS CIRCULATORS	(GCR) Manual trip of the reactor due to a problem with gas circulator (fatigue damage to the impeller)	GCR	PWR(40)	
A143	7584	REACTOR MANUAL SHUTDOWN FOR INSPECTION/ MAINTENANCE OF PRIMARY LOOP RECIRCULATION PUMP	(BWR) Manual shutdown to inspect/repair the recirculation pump (shaft vibrations)	BWR	BWR(36)	
A144	7585	REACTOR MANUAL SHUTDOWN DUE TO INCREASED READINGS OF MAIN STACK RADIATION MONITORS	(BWR) Manual shutdown to investigate problems with increased radiation in the moisture separator (leakage from one fuel assembly + deficiencies/leakage in 2 valves in the turbine building)	BWR	BWR(40)	
A145	7586	REACTOR MANUAL SHUTDOWN DUE TO LEAKAGE FROM THE VENT VALVE IN THE SEAL WATER INJECTION LINE OF THE REACTOR COOLANT PUMP	(PWR) Manual shutdown for repair. Cause – cracks due to erosion in the water injection line and vibrations in the control valve	PWR	PWR(36)	
A146	7587	AUTOMATIC REACTOR SCRAM BY ATWS, FOLLOWED BY A PRIMARY COOLANT LEAK IN THE REACTOR BUILDING CONTAINMENT	(PWR) Inappropriate action by the operator in the control room (trip of the turbine-driven feedwater pumps) led to test failure. The ATWS reactor protection signal was activated, starting up the auxiliary feedwater system (ASG) and generating an automatic reactor scram, which in turn caused the turbine to trip. Multiple failures – MFW pump control, mechanical weaknesses of valves, I&C logic of primary letdown valves	PWR	PWR(15)	
A147	7590	REACTOR TRIP ON GRID LOAD TRANSIENT	(PWR) Under-frequency operation of TG resulted in the turbine trip and generator trip. Damage to the turbine cross over piping is the non-opening of Moisture Separator Reheat valves # 1, following the actuation of Over Power Control. Hammering of the steam crossover piping resulting in the damage of expansion joints.	PWR	PWR(34)	
A148	7606	UNIT SCRAM BY RPS-1 PROTECTION DUE TO TRIPPING OF TWO OPERATING TURBINE GENERATORS	(WWER-440) Short circuit to earth in the stator winding of the circulation pump electric motor, generation of a spurious signal in the level measuring circuit of the high pressure heater causing closure of the turbine stop valves and turbine/reactor trip. There were additional problems with steam dump valves. Multiple independent failures	WWER	PWR(34)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A149	7617	UNIT DISCONNECTED FROM GRID BY ELECTRICAL PROTECTION SYSTEM OF AUXILIARY POWER TRANSFORMER WITH SUBSEQUENT TRIGGERING OF EMERGENCY PROTECTION SYSTEM	(WWER-440) Short circuit in the auxiliary transformer followed by protection signal, automatic switch to standby transformer, loss of 6 kV buses, and trip of the second (standby transformer (due to a damage to 6 kV bus caused by mechanical damage to aux transformer). Automatic reactor trip (AZ-1)	WWER	PWR(37)	
A150	7620	PICKERING RESPONSE TO A LOSS OF BULK ELECTRICAL SYSTEM	(PHWR) Loss of bulk electrical grid that affected most of Ontario and parts of the United States. Loss of Class IV power resulted in the loss of all high pressure emergency core injection for all units at the site for greater than 5 hours, which is greater than the 60 minutes assumed in the design. Loss of off site power also adversely impacted many other plant systems, and site response was complicated by equipment design or operational deficiencies.	PHWR	PWR(35)	
A151	7634	LOADING OF FUEL BUNDLES WITH SLIGHTLY DEFORMED ENDPLATES IN THE REACTOR CORE DUE TO MALFUNCTION OF FRESH FUEL TRANSFER SYSTEM AT KAIGA-1	(PHWR) Manual shutdown to investigate fuel loading problem.	PHWR	PWR(40)	
A152	7636	REACTOR SCRAM RESULTING FROM A LOSS OF NON-CLASS 1E UNINTERRUPTIBLE POWER SUPPLY AND ALERT CONDITION DECLARED AT LAGUNA VERDE NPP-1	(BWR) Electrical fault that resulted in transfer of uninterruptible power supply unit to alternative power source (no alarms generated). Re-transfer to original power source caused the fuse burnout and loss of power supply to control rod map display, position indication, source range monitoring system display, logic control of the flow control valves of the recirculation system, the control of FW turbine pump, the safety parameter display. Operator shuteddown the reactor manually.	BWR	BWR(36)	YES
A153	7636	REACTOR SCRAM RESULTING FROM A LOSS OF NON-CLASS 1E UNINTERRUPTIBLE POWER SUPPLY AND ALERT CONDITION DECLARED AT LAGUNA VERDE NPP-1	As above			
A154	7641	UNINTENDED REACTOR POWER RISE DUE TO TOTAL INCAPACITATION OF REACTOR REGULATION SYSTEM AT KAKRAPAR-1	(PHWR) Reactor regulating system failed due to a loss of power to all control rod drives. This led to slow increase of power (from 73% to 98%),. Reactor tripped on primary side high differential temperature across the SGs.	PHWR	PWR(39)	
A155	7650	BREACH OF SAFE OPERATION LIMITS FOR SG-1 LEVEL AND PRESSURE DUE TO BRU-A-1 ATMOSPHERIC RELIEF VALVE FAILURE TO CLOSE	(WWER-1000) Grid undervoltage conditions led to electrical transient which in addition to a failure of the excitation system led to TG disconnection from the grid. BRU-a and BRU-K opened. BRU-A failed to re-close (fault in limit switch). Limits on SG level and pressure, and pressurizer level were violated. Manual reactor scram	WWER	PWR(40)	
A156	7652	DUAL-UNIT SCRAM AT PEACH BOTTOM, UNITS 2 AND 3: NRC INFORMATION NOTICE 2004-15	(BWR) Dual unit facility lost offsite power, had a dual unit scram (automatic), and experienced other problems including the loss of a common emergency diesel generator. Unexpected E2 trip of one EDG during the cooling of the Unit 2 torus while other EDGs were supplying power to the emergency buses. This EDG shut down due to an engine protective trip initiated by low jacket water pressure.	BWR	PWR(35)	

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A157	7657	REACTOR SCRAM DUE TO FALSE ACTUATION OF MAIN GENERATOR PROTECTION WITH THE START OF ALL DGs	(WWER-440) Spurious TG protection. Load rejection caused overspeed, TG stop valves closed and reactor scram. Within 1.5 hr after the scram there was a loss of power fro 110 kV grid due to a spurious actuation of 6 kV bus protection. EDGs actuated. There were additional failures of two ESW pumps, EFW pump and LP ECCS pump to start. Cause - improper performance of load sequencer (automatics for undervoltage switch)	WWER	PWR(34)	
A158	7659	SCRAM DUE TO INSTRUMENTATION AND CONTROL TECHNICIAN MISTAKE BETWEEN UNITS	(WWER-440) Error of maintenance crew checking the SG level control at one unit led to a loss of power to level sensor in other unit (under operation) and in addition to independent failure of a relay in FW control circuit this cased turbine trip and reactor trip (on SG level high)	WWER	PWR(33)	
A159	7662	ANGRA-1 TRIP WITH SAFETY INJECTION DUE TO SPURIOUS ACTUATION OF CONTAINMENT HIGH PRESSURE SIGNAL CAUSED BY HAND-HELD RADIO	(PWR) Reactor trip. During the performance of containment pressure channel test a safety injection occurred as a result of spurious activation of containment high pressure signal caused by electromagnetic interference due to inappropriate use of portable radio close to containment pressure transmitters. Additional failures - RHR-2 pump had not automatically started due to circuit failure in the sequencer card, and feedwater isolation valve HV-1318 had not closed upon signal due to stuck stem.	PWR	PWR(39)	
A160	7668	PARTIAL INOPERABILITY OF REACTOR CONTROL AND LIMITATION SYSTEM CAUSED BY MULTIPLE SPURIOUS OPENING OF REACTOR TRIP BREAKERS	(WWER-1000) Reactor trip breakers affected by deficiencies in Reactor Trip System (multiple opening) led to loss of power control rod drives This led to power reduction, turbine trip and manual reactor trip.	WWER	PWR(33)	
A161	7678	UNIT SCRAM ON "ONE OUT OF TWO OPERATING REACTOR COOLANT PUMPS TRIPPED" SIGNAL OWING TO REDUCED TURBINE-DRIVEN FEEDWATER PUMP PERFORMANCE CAUSED BY A FEEDWATER PIPE LEAK	(WWER-1000) A leakage in non-isolable section of SG feedwater bypass line with abrupt steam pressure drop in the MSH and a reduction in the performance of turbine driven MFW pumps. Rapid drop of SG level led to RCPs trip followed by reactor trip. Cause pipe wall erosion and stress corrosion, spurious signal of steam header pressure to FW pumps due to penetration of water to the measurement channel terminal blocks	WWER	PWR(28)	YES
A162	7682	FAILURE OF PROTECTION RELAYS ON MAIN ELECTRICAL SUPPLY BOARD	(PWR) a short circuit on a non safety related motor lead not to the opening of its circuit breaker, the associated main electrical supply board switched off (on its overcurrent protection), leading to a loss of voltage on the associated equipments, including 2 primary pumps, thus inducing the reactor trip on "low primary pump speed signal"; Cause- inappropriate relay	PWR	PWR(37)	
A163	7687	SAFE SHUTDOWN OF KALPAKKAM-2 REACTOR FOLLOWING TSUNAMI STRIKE	(PHWR) Tsunami caused seawater entering to the intake tunnel and trip of the Condenser Cooling Water pumps. Operator tripped the turbine and reactor tripped on high pressure in PTH system. All CCW pumps and all process sea water pumps (submerged in seawater), except one, were unavailable. This pump provided cooling for the process water heat exchangers.	PHWR	PWR(30)	YES

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A164	7689	AUTOMATIC SHUTDOWN DUE TO FAST CLOSURE OF TURBINE CONTROL VALVE	(BWR) Closure of tG control valve during periodical testing led to automatic reactor scram (fault of the piston cylinder of the speed relay)	BWR	BWR(13)	
A165	7693	MANUAL SHUTDOWN DUE TO PRIMARY LOOP RECIRCULATION PUMP TRIP	(BWR) Primary loop recirculation pump B tripped on a signal of ‘serious fault in the inverter’, followed by a similar event for pump A (a signal “light fault of the inverter A”). Reactor shutdown manually to investigate problem. Causes – failures of electronic components of the inverter (pump B), reduced FW to the reactor and reduction of speed of pump A was a ‘normal’ action of protection system of pump A.	BWR	BWR(15)	
A166	7712	THREE-UNIT TRIP AND LOSS OF OFFSITE POWER AT PALO VERDE NUCLEAR GENERATING STATION: NRC INFORMATION NOTICE 2005-15	(PWR) Loss of off-site power (plant switchyard) originated with a fault across a degraded insulator on a 230 kV transmission line removing all sources of power to three nuclear units. The single-failure susceptibility of a transmission line protective system was the primary cause of the cascading blackout (the tripping scheme lacked redundancy). There was additional independent failure of one EDG (failed diode)	PWR	PWR(35)	YES*
A167	7715	PLANT TRIP AND LOSS OF PREFERRED AC POWER FROM INADEQUATE SWITCHYARD MAINTENANCE: INFORMATION NOTICE 2005-21	(PWR) Loss of power events as a result of inadequate preventive and corrective maintenance practices on switchyard breakers and current transformers. Event occurred during activities to reconfigure breakers in the 345 kV switchyard 2 phases in 3-phase breaker were opened the third failed to open in time causing a current imbalance in switchyard buses. Unnecessary plant trips and LOOP events could be reduced by following vendor recommendations with feedback from operating experience to determine the appropriate schedule and extent of maintenance. Similar observation made at other plants.	PWR	PWR(35)	YES*
A168	7721	UNIT SCRAM BY AZ-5 PROTECTION ON THE SIGNAL OF LO-LO LEVEL IN EMERGENCY CORE COOLING SYSTEM'S WATER ACCUMULATORS WITHOUT ACTUAL LEVEL DECREASE	(LWGR/RBMK) HE (opening of drain valve at the hydroaccumulator) led to a false signal of “low water level in HA” and reactor trip (AZ-5)	LWGR	PWR(39)	
A169	7727	LOSS OF POWER OF A 220 VAC SAFETY BUS BAR	(PHWR) Failure of monophasic inverter led to loss of 220V AC bus in uninterruptible (Class II) power supply system. In consequence the turbine control system and turbine bypass were lost, turbine trip and power reduction to 15%. One out of 3 logic channels of RPS was affected (tripped). Back-up computer and some monitoring equipment were also lost.	PHWR	PWR(37)	YES
A170	7731	INADVERTENT REACTOR TRIP AND PARTIAL SAFETY INJECTION ACTUATION DUE TO TIN WHISKER: NRC INFORMATION NOTICE 2005-25	(PWR) Unexpected safety injection actuation and reactor trip caused by a fault on a solid state protection system (SSPS) circuit card. The fault generated a false “low steam-line pressure signal”, bypassing the 2-out-of-3 SSPS logic and causing the A safety train actuation and reactor trip. The licensee examined the failed circuit card using a magnifying glass and found a microscopic tin filament (approximately 2 mm long). The filament created a bridge between the affected diode and the output trace on the card.	PWR	PWR(9)	YES*

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A171	7739	INADVERTENT ORDERS ISSUED FOR AUTOMATIC REACTOR SCRAM AND SAFETY INJECTION, RESULTING FROM WATER FLOODING THE FOUR SIP (PROCESS INSTRUMENTATION SYSTEM) ROOMS	(PWR) Start-up after refueling outage (low power). Flooding with hot water of instrumentation rooms containing I&C RPS system via electrical train ducts. Flood induced an inadvertent automatic reactor trip and initiation of safety injection. The cause was HE (failure to close the main feedwater drain valves together with blocking of one sump in the steam pipeline area).	PWR	PWR(9)	
A172	7767	PRIMARY/SECONDARY SYSTEM LEAKAGE RATE EXCEEDING 70 L/H IN A STEAM GENERATOR	(PWR) Primary to secondary system leakage that exceeded established limits (>70l/hr) was detected (detection by Nitrogen-16 measurement), unit shutdown manually.	PWR	PWR(26), PWR(40)	
A173	7772	PIPE RUPTURE OF SECONDARY SYSTEM	(PWR) Reactor trip and turbine trip on SG feedwater/steam flow mismatch trip signal due to a condensate water piping break in the turbine building (5 workers killed and 6 injured). Cause – erosion/corrosion induced by flow turbulence at the downstream of the orifice (pipe area not covered by the inspection programme)	PWR	PWR(28)	YES*
A174	7776	TEMPERATURE INCREASE OF CONTROL ROD DRIVE HOUSING 6/7 DURING RESTART OF PAKS UNIT 4	(WWER-440) Manual reactor trip caused by increased temperature of CR drive (due to leaking valve through which reactor coolant flowed from the upper block central venting line toward the makeup water system).	WWER	PWR(40)	
A175	7781	MANUAL REACTOR TRIP DUE TO LOSS OF ELECTRICAL DISTRIBUTION BOARD	(PWR) Manual trip to investigate electrical problem (trip of 3 inverters in uninterruptible power supply system). The cause – loose connection on the battery charger voltage divider card that cause inverter trip on over-voltage protection (dry joint on a soldered connection on the voltage divider printed circuit board.). Affected systems - Vital intrumentation AC and control AC	PWR	PWR(40)	YES
A176	7783	REACTOR MANUAL SHUTDOWN DUE TO INDICATION TROUBLE OF DRYWELL VACUUM BREAK VALVE POSITION	(BWR) Manual reactor trip to investigate problem with one of the drywell vacuum break valves (signal received that the valve is not fully closed). Problem appeared to be in the indication system (failed micro-switch)	BWR	BWR(36)	
A177	7785	REACTOR AUTOMATIC SHUTDOWN DUE TO LOW VACUUM OF CONDENSER	(BWR) Automatic reactor trip, turbine trip on low vacuum signal. The cause was failure of valve at the boiler supplying steam for turbine gland seal (unit was prepared for shutdown for periodic inspection) that let the air to the condenser.	BWR	BWR(8)	
A178	7788	LOSS OF 400 KV AND SUBSEQUENT FAILURE TO START EMERGENCY DIESEL GENERATORS IN SUB A AND SUB B	(BWR) Reactor scram due to disturbances in the off-site 400 kV switchyard. There were two transients in electrical power supply system: (1) overvoltage that led to a failure of 2 UPS in 220 V AC (needed for the operation of EDGs) and a low frequency transient that caused a disconnection of the off-site power to the safety buses. This led to a degradation of power supply function (2 EDGs not connected). It also caused several isolation signals and loss of information in the control room. Root cause - selectivity of the bus bar protections has not been adequate to protect the inverters	BWR	BWR(31)	YES
A179	7792	STEAM GENERATOR 10 % FEED WATER LINE BREAK IN SECONDARY CONTAINMENT OF REACTOR BUILDING	(PHWR) Rupture of feedwater 10% line (FW control branch) located inside the boiler room (secondary containment). Reactor tripped manually.	PHWR	PWR(40)	

#	IRS Number	Title	Description	NPP type	IE type (EPRI)	In-depth analysis ?
A180	7794	REACTOR POWER TRANSIENT DUE TO FAILURE OF LIQUID ZONE CONTROL SYSTEM	(PHWR) Failure of a valve in the reactor control system (subsystem based on the use of light water in the 4 zones in the core) led to decrease of the amount of light water in the zones and reactor power increase followed by reactor trip.	PHWR	PWR(39) PWR(2)	
A181	7795	SAFETY INJECTION AND REACTOR TRIP DUE TO AN UNSKILLFUL OPERATION DURING PLANT STARTUP	(PWR) Thermal-hydraulic transient in the main steam system due to operator error occurred that led to opening of the main steam safety valve (MSSV) and relief valve (PORV), followed by safety injection on low steam line pressure signal. A pipe segment silencer of the PORV was detached and became a flying missile (50 m) that collided with the side of RWST (permanent deformation of the wall).	PWR	PWR(29)	YES
A182	7797	TRIP DUE TO THE FAILURE OF PLANT DIGITAL CONTROL COMPUTER (DCC-X)	(PHWR) Failure of digital control computer (DCC_X) and incomplete transfer of the control function to the redundant computer (DCC-Y). Due to this failure SG level control was affected, the PRZ pressure increased. Due to a mismatch between the primary and secondary system power balance it led to reactor trip.	PHWR	PWR(19)	
A183	7800	REACTOR MANUAL SHUTDOWN DUE TO MALFUNCTION OF THE OUTLET VALVE OF MOTOR-DRIVEN REACTOR FEEDWATER PUMP-B	(BWR) Manual shutdown to inspect/resolve problem with the outlet valve (stem broken) of one of the reactor feedwater pumps.	BWR	BWR(36)	
A184	7801	ANGRA 1 TRIP DUE TO TOTAL LOSS OF OFFSITE POWER (LOPS) AND AUXILIARY FEEDWATER TURBINE-DRIVEN PUMP TRIP	(PWR) Reactor trip due to total loss of offsite power. Auxiliary feedwater pump trip occurred by overspeed which was caused by pressure increase in the steam line due to load rejection together with no steam dump to the condenser (not available due to loss of normal power)	PWR	PWR(35)	YES
A185	7806	LOSS OF SUPPLIES LEADING TO THE MANUAL SHUTDOWN OF REACTORS 1 & 2	(GCR) Electrical/ earth fault was detected in General Instrument (GI) electrical supplies system (one of the two redundant trains). Fault finding work programme was started but due to HE this led to loss of GI system and manual trip of reactors 1 & 2. The only deviation from normal plant response was overfilling of a deaerator due to lack of instrumentation. This event brings into question the adequacy of procedures and protection systems for electrical systems that require manual synchronization, especially where the ergonomics of control panels etc may be an issue.	GCR	PWR(40)	

* Indicate generic safety issues

Table A-2. *Query B search results (Search codes:1.2, 3.B, 3.C, 3.E, 3.H, 3.K; Incident between 1995-01-01 and 2006-11-15; items included in Table A-1 are omitted; total number of reports = 77).*

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B1	1522	POTENTIAL FOR LOSS OF AUTOMATIC ENGINEERED SAFETY FEATURES ACTUATION (NRC INFORMATION NOTICE 95-10, AND INFORMATION NOTICE 95-10, SUPPLEMENT 1)	Problems related to the solid state-protection system (SSPS) for Diablo Canyon and Salem NPPs during a steam line break (a steam jet could strike a junction box that contains terminations for non-safety input signals to SSPS. Induced electrical faults would cause a fuse to open and the loss of the automatic actuation function of engineered safety features (ESF). This common fuse failure would lead to loss of control power in one of the SSPS channel. If there is additional fault in second channel this may disable automatic actuation of the ESFs	PWR	PWR(29)	YES
B2	1589	LEAK FROM RESIUDAL HEAT REMOVAL SYSTEM AT A LOCATION PRECLUDING ISOLATION	Leakage from the primary circuit during the refueling shutdown (end of outage - RPV and PRZ closed, mid-loop operation water level) due to break of a vent nozzle of the RHR system common to both trains (~16 m3/h). This event required temporary repair and switching RHR off for a short time in order to change into SG cooling regime (increasing RCS parameters)	PWR	PWR(5) Shutdown	YES
B3	1621	CONSIDERATION OF VALVE MISPOSITIONING IN PRESSURIZED-WATER REACTORS (NRC GENERIC LETTER 89-10, SUPPLEMENT 7)	Generic issue related to mis-positioning of unlocked MOVs in safety related systems (removing earlier recommendations)	PWR		
B4	1622	MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT (NRC BULLETIN 96-02)	Generic issue related to moving heavy loads (dry storage casks) over safety related equipment during power operation. A cask drop could damage both isolation condensers and the torus, possibly creating an unisolable LOCA outside containment	BWR		YES
B5	7006	EXPLOSION AND FIRE ON HARTLEPOOL UNIT TRANSFORMER 1B RESULTING IN DECLARATION OF SITE INCIDENT	Explosion and fire on unit transformer that led to trip and disconnection from the grid	GCR	PWR(35)	
B6	7014	MAIN STEAM ISOLATION VALVE STEM FAILURE	MSIV inadvertently closed causing secondary pressure transient and initiation of SI and reactor trip. Cause – mechanical failure of the valve stem	PWR	PWR(18)	
B7	7038	MOTOR OPERATED VALVE PERFORMANCE ISSUES (NRC INFORMATION NOTICE 96-48)	Generic issue report related to MOV performance (including sizing of AS-powered motor actuators)			
B8	7086	HEAT TRANSPORT SYSTEM LEAK ON REACTOR FEEDER PIPE	Orderly shutdown due to a leakage from a reactor channel outlet feeder pipe without special safety system action being invoked or required	PHWR	PWR(5)	
B9	7105	IMPROPER ELECTRICAL GROUNDING RESULTS IN SIMULTANEOUS FIRES IN THE CONTROL ROOM AND THE SAFE-SHUTDOWN EQUIPMENT ROOM (NRC Information Notice 97-01)	A single electrical fault caused simultaneous fires in the control room and the Train B dc equipment room which supports alternative post-fire safe shutdown capability in the event of a control room fire. This electrical design error created fire vulnerability in two separate areas of the plant. The fire could have resulted in operational challenges which are outside of the plant's design basis and the scope of the NRC fire protection regulations (10 CFR 50.48). This vulnerability was caused by the inadequate design of	PWR	PWR(41)	YES

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
			the grounding circuitry from the electrical power supplies, which have been in service since the original construction			
B10	7107	LICENSEE RESPONSE TO INDICATIONS OF TAMPERING, VANDALISM, OR MALICIOUS MISCHIEF (NRC Information Notice 96-71)	Generic issue report on cases of tampering, vandalism or malicious mischief with respect to safety related equipment			
B11	7119	PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1995, A STATUS REPORT (NUREG/CR 4674 Vol. 23)	Generic report on the results of the review and evaluation of 1995 operational experience data by the Nuclear Regulatory Commission's ongoing Accident Sequence Precursor (ASP) Program. The most relevant precursors include (1) failure of pressurized power-operated relief valves, (2) failures of multiple stages of a reactor coolant pump seal, and (3) the unavailability of shutdown cooling. No details can be found because the report provided to IRS database is not complete			
B12	7129	IMPAIRMENT OF SHUTDOWN SYSTEM NO. 2 (SDS2) DUE TO MAINTENANCE ERROR - WRONG UNIT CALIBRATED	HE during calibration of SDS2 ‘regional overpower’ (ROP) detectors (wrong unit selected) led to a degradation of systems required to control reactivity. Total unavailability of SDS2 for 37 minutes	PHWR		YES
B13	7130	ERROR FOUND IN THE NEUTRON OVERPOWER (NOP) REFERENCE MAP USED IN A SAFETY ANALYSIS	Generic issue report regarding discrepancies in the channel power maps in the SAR, safety analysis and utility fuel management programme. This had impact on Regional Overpower system and SDS protection system. Negligible effect of the discrepancy on required NOP trip setpoints	PHWR		
B14	7164	FALSE SIGNAL FOR CLOSING FAST ACTING MAIN STEAM ISOLATION VALVE 4 AT KOZLODUY NPP UNIT III ON 10 SEPTEMBER 1997	(WWER-440) Power reduced from 95% to 75% due to a false signal for closing FAMSIV No. 4 (SG line connection with the MSH) leading to main circulation pump (MCP) No. 4 trip. The direct cause of the event was the presence of non-isolated cable in the FAMSIV No. 4 limit switch switchboard which caused short circuit within the switchboard.	PWR	PWR(17)	
B15	7166	FUEL FAILURE DURING CYCLE 6 OF ANGRA 1 NUCLEAR POWER PLANT	8 out of 40 fuel assemblies in batch G (121 total in the core) were found in failed state. Direct cause – flow induced vibration and rod fretting wear.	PWR		
B16	7176	FIRE ENDURANCE TEST RESULTS FOR ELECTRICAL RACEWAY FIRE BARRIER SYSTEMS CONSTRUCTED FROM 3M COMPANY INTERAM FIRE BARRIER MATERIALS (NRC Information Notice 95-52, Supplement 1)	Testing fire barriers (Interam E54A material) of electrical raceways was not fully successful			
B17	7192	WATER HAMMER EVENTS SINCE 1991 (NRC INFORMATION NOTICE 91-50, SUPPLEMENT 1)	Generic issue report. Water hammer events were observed in several plants PWR and BWR plants in various systems mostly in secondary side systems			YES
B18	7202	REACTOR MANUAL SHUTDOWN DUE TO A MALFUNCTIONING OF A CONTROL ROD	Manual shutdown to investigate problem with CR drive system (discovered during a periodic test)	BWR	BWR(36)	

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B19	7203	REACTOR MANUAL SHUTDOWN DUE TO A MALFUNCTIONING OF A CONTROL ROD	Manual shutdown to investigate problem with CR drive system (one of the CRs could not be withdrawn during control rod pattern adjustment)	BWR	BWR(36)	
B20	7209	INADVERTENT REACTOR BUILDING CONTAINMENT SPRAYING	Containment spray inadvertent operation and containment isolation occurred during bimonthly surveillance test of CSS and CCS MOVs due to a failure of electronic component in a relay card of the RPS. Additional failure was ineffective resetting of CNT isolation signal by the operator, which led to RCS bearing temperature increase and automatic reactor scram. Multiple failures	PWR	PWR(31)	
B21	7214	POTENTIAL DEFICIENCY OF ELECTRICAL CABLE/CONNECTION SYSTEMS (NRC INFORMATION NOTICE 98-21)	Generic NRC report. Thermal and radiation ageing followed by LOCA simulation test. Some deficiencies were found in the steam exposure tests (post-LOCA conditions). Potential for CCF. In certain nuclear power plant circuit applications, degradation or failure of electrical cable/connection systems during and/or following design-basis events can significantly affect the functional performance of safety-related equipment			YES
B22	7215	SHAFT BINDING IN GENERAL ELECTRIC TYPE SBM CONTROL SWITCHES (NRC INFORMATION NOTICE 98-19)	Generic problems of safety interest. The problem is related to control board switches to actuate safety-related pump motors, motor- operated valves, and circuit breakers. Binding that prevents resetting after use due to material/design problem (post-mold cure shrinkage)			YES
B23	7239	BULK ROD ASSEMBLY LOADED IN ERROR INTO REACTOR 8 CHANNEL PQ90 INSTEAD OF SENSOR ROD ASSEMBLY	Degradation of systems required to control reactivity. No reactor shutdown.	GCR		
B24	7257	MOTOR-OPERATED VALVE PERFORMANCE ISSUES (NRC INFORMATION NOTICE 96-48, SUPPLMENT 1)	Generic issue related to sizing AC-powered motor actuators for MOVs (Limitorque Co.). Previous relaxation of the sizing criteria was retracted			YES
B25	7272	UNAVAILABILITY OF THE ECCS DUE TO RUPTURE OF A PIPE OF THE SG BLOWDOWN SYSTEM DURING THE ANNUAL OUTAGE OF UNIT 1	Event was during planned refueling outage (RPV open and connected with SFP). A leak in the SG-3 blow-down system led to penetration of the chemical treatment solution into ECCS tank (insufficient capacity of the SG room drain system) and need for full draining of the tank to perform cleaning. Degradation of an essential safety system; potential for CCF.	PWR		YES*
B26	7281	AGEING OF CABLES IN THE STEAM GENERATOR SPACE OF LOVIISA REACTORS	Generic issue. Deterioration of thermal insulation of the piping in the SG box and increased temperature level led to increased rate of cable ageing. The effect on safety was found small. However, potential CCF mechanism is indicated.	PWR/ WWER		YES
B27	7303	ERRONEOUS SAFETY SYSTEM STATUS CONTROL AFTER OUTAGE	Generic issue. Safety components erroneously left inoperable after an outage (9 events from NPPs in Sweden). Problems due to complex work situation, time pressure, insufficient procedures, etc.	PWR/ BWR		YES
B28	7309	REACTOR MANUAL SHUTDOWN DUE TO THE FAILURE OF THE PRIMARY LOOP RECIRCULATION PUMP (A) MECHANICAL SEAL	Manual shutdown to inspect problem with leakage from the mechanical seal of the primary loop recirculation pump.	BWR	BWR(36)	

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B29	7332	FRACTURE OF INJECTION VALVE SPINDLE OF LOW PRESSURE CORE SPRAY SYSTEM OF TOKAI NO. 2 POWER STATION	Degradation of safety-related system (an injection valve in the LP Core Spray system)	BWR		
B30	7339	RESOLUTION OF GENERIC ISSUE 145, ACTIONS TO REDUCE COMMON-CAUSE FAILURES (NRC REGULATORY ISSUE SUMMARY 99-03)	Generic report on CCFs identified in LER reports (during 15-year period 1980-95)	PWR/ BWR		YES
B31	7342	PARTIAL LOSS OF SAFEGUARD SYSTEMS AS A RESULT OF EXTERNAL FLOODING	Preliminary report (see below for final)	PWR		
B32	7344	ANGRA 1 REACTOR TRIP DUE TO LOSS OF THE PREFERRED OFF SITE POWER 138kV AND INVERTER BOP-2 FAILURE	Loss of two safety buses and failures in the inverter BOP-2 and in the generator protection circuit led to loss of MFW (one pump in corrective maintenance, one lost due to undervoltage, the third failed due to BOP-2 inverter failure), followed by a turbine and reactor trip and total loss of off-site power (generator load break switch did not open after 5 seconds of the transient). Multiple failures but initiated by electrical system disturbances (dependencies)	PWR	PWR(16)	YES
B33	7352	DEGRADATION OF PIPE COMPONENTS ON RECIRCULATION SECTIONS OF CONTAINMENT SPRAY SYSTEM	Degradation of the containment spray system (distortion of expansion bellows in both trains). Problem caused by an alignment error. Potential to pressurize of one train when the pump of the other train is being tested. This dependency was due to inadequate design/modification and badly assembled expansion joints.			YES
B34	7361	POTENTIAL FIRE HAZARD IN THE USE OF POLYALPHAOLEFIN IN TESTING OF AIR FILTERS (NRC INFORMATION NOTICE 99-34)	Use of flammable aerosol in testing a high-efficiency particulate air (HEPA) filter led to potential fire event (flame generation)		PWR(41)	YES*
B35	7397	CRACKS IN SAFE END WELD AREA OF OUTLET REACTOR NOZZLE IN RINGHALS 4 NPP UNIT	Cracks in the RCS due to stress corrosion. No IE.	PWR		
B36	7404	AUTOMATIC TURBINE CONTROL SYSTEM DEGRADATION BECAUSE OF INCOMPATIBLE ELEMENTS USE DURING MAINTENANCE	Natural ageing of capacitors in the turbine automatic regulation/protection system (on-line memory module) led to blockage of relating channels of the system (different characteristics of capacitors in the same cubicles.	LWGR		YES
B37	7435	REGULATORY EFFECTIVENESS OF THE STATION BLACKOUT RULE	Review of the station blackout (SBO) rule and its practical effectiveness			
B38	7436	EVALUATION OF AIR-OPERATED VALVES AT U.S. LIGHT-WATER REACTORS (NRC NUREG-1275, VOLUME 13)	A study of air operated valves; review of current requirements and operating experience.			YES
B39	7449	HEAT TRANSPORT FEED PUMP WEAR RING FAILURES	Degradation of safety-related system. Feed valve in HT system stuck open. Event occurred during unit restart following the completion of repairs to a leak from the secondary side of the SG. Cause – stress corrosion cracking. No IE	PHWR		

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B40	7452	SECOND SHUTDOWN SYSTEM LIQUID INJECTION SYSTEM HELIUM SUPPLY CHECK VALVE OPERABILITY ISSUE NOT RECOGNIZED	Degradation of safety-related system (SDS2). Check valve in helium system does not prevent reverse gas flow that could lead to a loss of SDS2 in a seismic event	PHWR		
B41	7480	STRESS CORROSION CRACKING IN AUSTENITIC STAINLESS STEEL COMPONENTS	Degradation of safety-related system. Stress corrosion cracking on the SI piping (austenitic steel 304L) due to environmental site-related conditions (high marine salt concentration). Initially underestimated in the design.	PWR		YES*
B42	7485	REACTOR SCRAM FOLLOWING UNIT DISCONNECTION FROM GRID DUE TO INCORRECT MANIPULATIONS IN EXTERNAL SWITCHING STATION	Off-site power (220 kV) line lost due to HE that led to turbine trip (by frequency protection system)	PWR WWER	PWR(33)	
B43	7495	LOSS OF SAFETY FUNCTION ON THE ENHANCED SHUTDOWN (ESD) SYSTEM DUE TO A WIRING DESIGN ERROR	Degradation of safety-related system (Enhanced Shutdown ESD). Loss of functionality of 20 ESD control rods due to a wiring design error (introduced during a minor modification to the overall system). Deficiency not recognized since the modification (revealed during routine maintenance)	GCR		YES
B44	7498	SAFETY ASSESSMENT OF PRIMARY COOLANT UNISOLABLE LEAK INCIDENTS CAUSED BY THERMAL FATIGUE	Generic report on unisolable primary coolant leakages fro auxiliary piping connected to RCS induced by cracking caused by fatigue. 9 events from international experience reviewed by IPSN-GRS working group.	PWR		YES*
B45	7502	DISCOVERY OF SHORTING LINKS WHICH HAD BEEN LEFT IN PLACE IN ERROR AND DEFEATED THE DOUBLE POLE/DOUBLE THROW TRIP SYSTEM ON THE FUELLING MACHINE	Deficiencies were identified during the testing in the protection system of refueling machine that protects against dropping irradiated fuel. Shorting links (across two pairs of terminals in one of the control system cabinets on the machine) were found, which defeated both channels of the relevant sub-system. This was a HE made during re-commissioning checks.	GCR		YES
B46	7510	REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY	Generic report on the integrity of the RCS boundary (due to corrosion, erosion, fouling); related regulatory requirements and licensee's inspection practices	PWR		
B47	7510	RECENT EXPERIENCE WITH DEGRADATION OF REACTOR PRESSURE VESSEL HEAD	Generic report – findings from inspection (see above)	PWR		
B48	7510	POSSIBLE INDICATORS OF ONGOING REACTOR PRESSURE VESSEL HEAD DEGRADATION	Generic report – applicable indicators to alert operators to RCS boundary degradation (see above)	PWR		
B49	7534	CHOKING OF SCREENS AT CONFINEMENT SPRAY PUMP SUCTION	Screens at the inlet of spray pumps (remnant of post-installation cleaning operation – not removed due to a HE) led to a increased pressure drop during the test	PWR WWER		YES
B50	7540	SERIOUS FUEL CLADDING FAILURES AT CATTENOM 3	92 fuel rods for the 28 leaking assemblies failed due to a vibratory phenomenon ("fretting"). Failures identified trough increased activity of the coolant and examined during refueling outage.	PWR		

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B51	7547	MULTI-UNIT POWERHOUSE EQUIPMENT DAMAGE RESULTING FROM OPERATION OF EMERGENCY POWERHOUSE VENTING SYSTEM	Generator hydrogen system failure (oil tank level control) occurred that led to loss of seal. In order to mitigate the build-up of hydrogen in the powerhouse emergency venting (PHEV) was initiated that led to frizzing of instrument lines in the vicinity of PHEV on all units. Automatic load reduction. No IE	PHWR		YES
B52	7549	CRACKING OF A CONTROL ROD DRIVE MECHANISM PROTECTION PIPE	Transgranular stress corrosion cracking was identified during walk-down inspection (cristalized boric acid on the surface of the insulation shield box of the temperature measurement device of the CRD mechanism protection pipe).	PWR		YES
B53	7557	RECENT EXPERIENCE WITH REACTOR COOLANT SYSTEM LEAKAGE AND BORIC ACID CORROSION (NRC INFORMATION NOTICE 2003-02)	Degradation of barriers. Boric acid corrosion and RCS leakages (2 plants); in one case a degradation of the RPV head	PWR		YES
B54	7569	PIPE RUPTURE INSIDE CONTAINMENT DUE TO RADIOLYSIS GASES EXPLOSION	Explosion of radiolysis gases accumulated in the RPV head spray (RPVHS) line caused leakage of steam inside containment. Unforeseen interaction between systems. Manual load reduction. No IE	BWR		YES
B55	7576	LEAK AT A REACTOR COOLANT PUMP THERMAL BARRIER	After installation of a new-design thermal barriers on the RCP a leakage were detected. Problem was due to an incorrect evaluation of the force on the screws used to assemble flange and the housing of this thermal barrier. Degraded reactor coolant boundary. No IE	PWR		
B56	7583	INCREASED RISK TO KOEBERG NPP UNIT 2 AS A RESULT OF DEGRADED 6,6 KV BREAKERS AND FAILURE OF TWO DIESEL GENERATORS ON DEMAND	Several failures some of them of CCF nature (breakers – inappropriate roller hardness). Replacement of failed breakers required DG operation. Diesel failures independent events (causes – faulty relay and administrative controls, respectively).	PWR		YES
B57	7587	AUTOMATIC REACTOR SCRAM BY ATWS, FOLLOWED BY A PRIMARY COOLANT LEAK IN THE REACTOR BUILDING CONTAINMENT	Test conducted for the turbine-driven feedwater pumps of the SG feedwater system failed due to a HE that led to automatic reactor scram (on ATWS signal). There were additional independent failures (the condenser bypass relief valve failed open, valve in the coolant letdown system, vent valves in the excess letdown heat exchanger left open after the containment test) that led to overcooling of the RCS, low level in the pressurizer and subsequent overfilling of the pressurizer (due to operator actions and failure of the letdown system), and release of primary coolant to the containment. Accident precursor.	PWR	PWR(21)	
B58	7593	LOSS OF COOLANT FROM REACTOR COOLANT SYSTEM ISOLABLE PART DUE TO LEAKING VALVES	RCS leakage through a valve at the auxiliary system (spent fuel storage pool cooling and cleanup). Degraded RCS boundary. No trip.	PWR WWER	PWR(5)	
B59	7594	RUPTURE OF REACTOR COOLANT SYSTEM DRAIN PIPE DURING PRESSURE TEST AT 16.8 MPA	Rupture of the drain line of the hot leg loop during the pressure test at unit outage due to fatigue loading. The most probable cause of the pipe break was the occurrence of hydraulic shocks due to either periodic opening and closing of the relief valve in the drain line, or by the flow of steam-water mixture through the discharge piping from the relief valve open.	PWR	PWR(5)	

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B60	7600	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS - NRC BULLETIN 2003-01	Generic report on the results of NRC research regarding potential susceptibility of recirculation sump screens to debris blockage in the event of high energy pipe breaks requiring recirculation of the ECCS and CSS	PWR		YES
B61	7603	UNAVAILABILITY OF LPSI FUNCTION DUE TO SIS 61 AND 62 VP VALVE CLOSING DURING DAMPIERRE UNIT 2 SHUTDOWN	Procedural error led to the operation of the unit in the cold shutdown state (26 bars, 40 C) with SI unavailable (isolated). Degradation of a relevant safety system. No IE.	PWR		
B62	7642	DELAY IN IDENTIFYING REACTOR 8 AUXILIARY GUARDLINE TRIP AND SAFETY RODS TRIP INTO THE CORE	Deficiencies in the management of reactivity control in the CR when the unit is shutdown (errors in the maintenance of reactor guard system and safety rods tripped into the core not detected)	GCR		
B63	7664	UNAVAILABILITY OF AUTOMATIC MAKEUP TO THE REACTOR COOLANT SYSTEM DURING TRANSITION TO MID-LOOP OPERATION	Degradation of heat removal capability. A precursor event. Automatic make-up system was not available due to relay module withdrawn from the cabinet.	PWR		
B64	7722	RISK OF CONTAINMENT SUMP SCREEN BLOCKAGE	IRSN generic report on the risk of containment sump screen blockage jeopardizing post-LOCA core cooling capability.			YES
B65	7733	SAFE SHUTDOWN POTENTIALLY CHALLENGED BY UNANALYZED INTERNAL FLOODING EVENTS AND INADEQUATE DESIGN (ML051440302): LICENSEE EVENT REPORT 305-2005-004 (KEWAUNEE)	Deficiencies in the protection of relevant safety equipment from the postulated internal flooding events (based on LER) discovered during plant refueling outage.	PWR		YES
B66	7733	SAFE SHUTDOWN POTENTIALLY CHALLENGED BY UNANALYZED INTERNAL FLOODING EVENTS AND INADEQUATE DESIGN: NRC INFORMATION NOTICE 2005-30	As above	PWR		
B67	7740	COMPLIANCE DEVIATION OF K1 CONNECTION BOXES	An isolation faults in the electrical junction boxes of containment isolation valves of the nuclear sampling system caused by inadvertent spraying from a leak on the main FW system. Investigations showed that there are many similar faults (electrical cables with notched insulating sheets, exposed wires, shrunk insulation sheets, etc.) in several plants. Potential consequences for safety in case of degraded atmosphere conditions in the containment.	PWR		YES
B68	7741	DEFECTIVE OPERATION OF 6.6 KV ELECTRICAL CIRCUIT BREAKERS	Increased rate of failures of 6.6 kV circuit breakers (failure to latch when called upon to close). Circuit breakers with little regular movement found vulnerable to gumming effect due to grease ageing.	PWR		YES
B69	7745	UNSOLVED LONG-TERM INDICATION OF EXCESSIVE PRESSURE ON THE MAIN COOLANT PUMP SERVICE DECK A301/1,2	Exceeding technical specification limits (increased pressure in the main coolant service deck) due to a HE (the outlet of circulation system KLA 10 of HVAC system interconnected with the main coolant pump service deck via common air duct of VHAC system, through open flaps and non hermetic cable penetrations)	PWR WWER		

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
B70	7748	INSUFFICIENT WATER LEVEL IN SUMP 3 CSS 004 BAFOR RECIRCULATION ON CSS SPRAY LINE	Significant degradation of safety function due to reduced water levels in the ECCS/CSS sumps (for 64 days) which would lead to a vapour lock effect in the suction valve of the sump and unavailability of ECCS/CSS system. The most probable cause of the reduced water level is a migration of a pocket of air trapped in the "check valve" pipe section towards the sump. Other causes of the problem are possible including alignment errors during maintenance.	PWR		YES
B71	7763	USE OF GALVANIZED SUPPORTS AND CABLE TRAYS WITH MEGGITT SI 2400 STAINLESS- STEEL-JACKETED ELECTRICAL CABLES: NRC INFORMATION NOTICE 2006-02	Stainless-steel-jacketed electrical cables found vulnerable to degradation under fire conditions when in contact with cable tray or support made of galvanized material (results of fire tests). Stainless steel trays are recommended. Degradation of plant control. Potential for CCF			YES
B72	7765	WATER LEAKAGE FROM CONTAINMENT DUE TO DEGRADATION OF THE METAL LINER IN RINGHALS 2 NPP	Manual shutdown to investigate the containment leakage problem. The problem was related to the leaktighness of the liner ("toroid plates" used to join the cylindrical liner wall to the liner in the ground concrete plate). Cause of the problem – corrosion due to unfavourable environment. Degradation of the containment system.	PWR	PWR(40)	
B73	7767	PRIMARY/SECONDARY SYSTEM LEAKAGE RATE EXCEEDING 70 L/H IN A STEAM GENERATOR	Manual shutdown of the reactor in response to increased leakage rate (detected by N-16 measurement)	PWR	PWR(26)	
B74	7778	POSSIBLE DEFECT IN BUSSMANN KWN-R AND KTN-R FUSES, NRC INFORMATION NOTICE 2006-05	Generic issue report on the subject of failures of fuses (Bussmann KWN-R and KTN-R type). The cause was a poor solder joints that resulted in a loss of electrical continuity of the fuse while in service. Potential for CCF.			YES
B75	7780	DISCOVERY OF A FOREIGN MATERIAL IN THE PIPING OF THE CONTAINMENT SPRAY SYSTEM	A foreign material discovered during a refueling outage in a containment spray system pipe. It could have led to the partial or total unavailability of the function in an accident conditions through the loss of this CSS train	PWR		
B76	7801	ANGRA 1 TRIP DUE TO TOTAL LOSS OF OFFSITE POWER (LOPS) AND AUXILIARY FEEDWATER TURBINE-DRIVEN PUMP TRIP	Reactor trip due to total loss of offsite power. Additional abnormalities including auxiliary feedwater turbine-driven pump (tripped by overspeed) and rupture of the LP turbine disc. These failures were induced by the transient but occurred due to independent deficiencies	PWR	PWR(35)	
B77	7802	UNAUTHORIZED BREACH OF CONTAINMENT DURING GROUP B DOUSING VALVES CONTROL AIR PNEUMATIC PILOT VALVES TEST	Multiple opening of the containment envelope (of very short duration) during testing of the pneumatic valves in the dousing system (connection through the instrument lines). Containment isolation problem. No IE	PHWR		

* Indicate generic safety issues

Table A-3. *Query C search results (based on key words and manual screening; date of incidents between 1995-01-01 and 2006-11-15; 34 selected events out of total 200 reports identified using key-word search)*

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
C1	1516	SUSCEPTIBILITY OF CONTAINMENT SUMP RECIRCULATION GATE VALVES TO PRESSURE LOCKING (NRC INFORMATION NOTICE 95-14)	Potential susceptibility of containment sump recirculation gate valves to pressure locking identified during a re-evaluation of valves weaknesses. Problem is caused by the pressurization of the water solid valve bonnet (due to a leakage of water from the pump side) during LOCA conditions that would prevent the valve from opening.	PWR		YES
C2	1525	POTENTIAL PRESSURE-LOCKING OF SAFETY-RELATED POWER-OPERATED GATE VALVES (NRC INFORMATION 95-18)	Problem similar to 1516 but in another plant	PWR		YES
C3	1599	SUSCEPTIBILITY OF LOW-PRESSURE COOLANT INJECTION AND CORE SPRAY INJECTION VALVES TO PRESSURE LOCKING (NRC INFORMATION NOTICE 95-30)	Several failures experienced (identified during testing) with gate valves in the LP coolant injection and core spray injection systems. The problems were caused by pressure locking (shearing of the motor shaft and the pinion key, and overheating of the motor)	BWR		YES
C4	1601	INADEQUATE OFFSITE POWER SYSTEM VOLTAGES DURING DESIGN-BASIS EVENTS (NRC INFORMATION NOTICE 95-37)	Generic report. Shortcomings in the plant site voltage regulation may lead to a separation of the ESF loads from offsite power and re-sequencing of the ESF loads onto the DGs. This may occur when there is a heavy loading of the start up transformer (operation with the switchyard voltage in the lower two-thirds of its expected operating range). Following a main generator or turbine trip that would accompany the LOCA may lead to a degradation of essential support systems. Several licensees have reported similar deficiencies in their electrical power distribution systems			YES
C5	6392	DEENERGIZATION OF 6 kV ESSENTIAL BUS BY UNDUE ACTION OF THE EARTH FAULT PROTECTION AT KURSK NPP	The 6 kV reliable power supply distribution panels were deenergized twice owing to spurious action of the earth-fault protection system on the sectionalizing circuit-breakers. Direct cause were a damaged motor stator winding of the fire-fighting pump (unit 4) and a short-circuit in the terminal box of the motor of the condensate pump (Unit 3). Event occurred at full power. In the latter case there was an IE and a degradation of MFW system (minor).	WWER	PWR(35)	YES
C6	6397	UNAVAILABILITY OF THE TRAINS OF THE EMERGENCY POWER SUPPLY SYSTEM'S ESSENTIAL COMPONENTS OWING TO FAILURES OF THE REVERSIBLE MOTORGENERATORS AT NOVOVORONEZH NPP	Several events of partial loss of in-house power supply due to failures of reversible motor generators. Causes – expired lifetime and poor manufacturing quality	WWER	PWR(37)	
C7	7013	OPERATING EVENTS WITH INAPPROPRIATE BYPASS OR DEFEAT OF ENGINNERED SAFETY FEATURES(AEOD/E95	Generic report on the results of NRC evaluation of operator’s actions in the control of engineered safety features equipment (ESF). Some deficiencies in the related procedures, operator’s knowledge of the procedures and poor watch-standing practices (communication, shift turnover, control board walkdown, verification of automatic actions, and response to alarms), which affect operator’s decision to bypass, defeat or turn off a safety system, are indicated			YES*

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
C8	7135	POTENTIAL NITROGEN ACCUMULATION RESULTING FROM BACKLEAKAGE FROM SAFETY INJECTION TANKS (NRC Information Notice 97-40)	Generic report on the potential effect of nitrogen accumulation in interfacing systems from safety injection tanks (SITs) due to a leakage of nitrogen-saturated water. Gas voiding in LPI/RHR systems may lead to waterhammer effects and in consequence to waterhammer-induced pressure locking (see report 1599)	PWR		YES
C9	7155	POTENTIAL FOR WATER HAMMER DURING RESTART OF RESIDUAL HEAT REMOVAL PUMPS (NRC Information Notice 87-10, Supplement 1)	Generic report on the potential for waterhammer in the RHR system in BWRs during a design basis LOCA coincident with a loss of off-site power (LOOP) if the RHR is aligned in the suppression pool cooling (SPC) mode of operation. Experience shows that RHR in SPC mode is frequently used in normal operation to remove heat from leaking safety relief valves. This increases the probability that response to a LOCA will require re-alignment of RHR from SPC to LP safety injection mode which therefore increases the possibility of waterhammer damage. This is potential dependency mechanism leading to CCI events	BWR		YES
C10	7228	MANUAL TRIP OF AUXILIARY FEEDWATER PUMPS DURING AUTOMATIC OPERATION	Event occurred during re-start of the plant after refueling outage at low power level (reduction of power and shutting down the plant to identify a source of leakage in the containment). Auxiliary feedwater pumps which are used at this situation were turned-off by operators (HE - overriding of the procedure) and the valve at suction line erroneously closed (HE). Cavitation and waterhammer led to a damage of both AFW pumps. Reactor tripped on the signal of “isolation of the reactor building area” initiated due to a leakage from the pump seals. IE and loss of all AFW trains.	BWR	BWR(34)	YES
C11	7254	STEM BINDING IN TURBINE GOVERNOR VALVES IN REACTOR CORE ISOLATION COOLING (RCIC) AND AUXILIARY FEEDWATER (AFW) SYSTEMS (NRC INFORMATION NOTICE 98-24)	Generic report on the design deficiency that prevented operation of safety-related system (RCIC and AFW (stem binding of the RCIC turbine governor valve due to thermal expansion of the stem and carbon steel washers and carbon spacers)	BWR		YES
C12	7259	LOSS OF INVENTORY FROM SAFETY-RELATED, CLOSED-LOOP COOLING WATER SYSTEMS (NRC INFORMATION NOTICE 98-25)	Generic report on the potential inoperability of safety related, closed-loop cooling water systems due to loss of inventory from excessive leakage combined with the absence of a reliable and timely annunciation and/or make-up system (typically, the make up system path from demineralized water system is not seismically qualified and pumps not powered from essential EPS (class 1E) system)		PWR(31) PWR(32)	YES*
C13	7285	PLANT SHUTDOWN CAUSED BY THE CRACK OF THE ESSENTIAL SERVICE WATER SYSTEM CONCRETE PIPE	The event occurred at power during the test of essential SW system (standby pump start/actuation relays). A leakage of water from the broken SWS pipe (in standby train B). Plant shutdown manually by TS LCO requirements	PWR	PWR(40)	
C14	7305	DE-ENERGIZATION OF THE 6 KV 1RB IN-HOUSE BUS DUE TO OPERATOR ERROR	Event occurred at power (power rising regime) when checking the excitation system parameters of turbine generators. Failure of a circuit breaker and a HE led to a loss ofpower on two 6 kV buses with starting up the dedicated DG-2.	LWGR RBMK	PWR(37)	
C15	7320	LUBRICATION-RELATED COMMON MODE FAILURES	Generic report on significant safety-related incidents associated with the unsatisfactory use of lubricants (4 plants) in mechanical (pumps, valves) and electrical (servo-motor, circuit breaker) components	PWR		YES

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
C16	7381	FAILURE OF COOLANT TEMPERATURE REGULATING VALVES OF EMERGENCY DIESEL GENERATORS	Generic report on the design deficiency introduced as a result of modification. There were several failures of a control valve in the DG cooling system that led to DG trip on high temperature in the cooling system signal. The valve had been modified to resolve another problem observed in the past with thermostatic valves. No IE	PWR		YES
C17	7410	NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER (NRC INFORMATION NOTICE 2000-14)	The event occurred at power. It involve phase-to-phase electrical fault in 12 kV non-Class 1E electrical bus duct from the unit auxiliary transformer to the switchboard that supply RCP and CW pumps, that led to turbine and reactor trip and started a fire, which affected another non-Class 1E 4 kV bus from the startup transformer located above the original failure.	PWR	PWR(33) PWR(41)	YES
C18	7411	LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION (NRC INFORMATION NOTICE 95-03, SUPPLEMENT 2)	Generic report regarding the potential for adverse effect on accident mitigation capability (ECCS) under LOCA accident in a hot, pressurized, shutdown conditions due to deficiency of a procedure. It directs the alignment of the suction of the operating charging pump to the common ECCS suction header, potentially exposing it to hot reactor coolant and rendering it inoperable. The same applies to the second charging pump and SI pumps if they need to be used to replace the inoperable ones.	PWR		YES
C19	7433	SHORT-TERM INOPERABILITY OF ALL FOUR EDGS AT A UNIT AT FULL POWER	Significant degradation of safety function. Degraded conditions involved all 4 DG for 4 hrs (associated breakers left in a 'revision' position i.e. DGs unavailable)	PWR WWER		
C20	7438	DOUBLE REACTOR TRIP ON LOSS OF GENERAL INSTRUMENT SUPPLIES	Manual trip of the unit 1 to address a problem with a stack fuel element and a loss of General Instrument supplies for the site and loss of Data Processor and the majority of central CR indications. Reactor 2 desk engineer confused with the remaining indications made some mistakes that led to the automatic tripping of reactor 2.	GCR	PWR(40)	
C21	7441	POTENTIAL LOSS OF REDUNDANT SAFETY- RELATED EQUIPMENT BECAUSE OF THE LACK OF HIGH-ENERGY LINE BREAK BARRIERS (NRC INFORMATION NOTICE 2000-20)	Generic NRC report on the potential to failure of the safety related systems due to a high energy line break (due to harsh environment of a HELB). Two event scenarios are mentioned in which there were deficiencies in the existing barriers against this IE. It involve CCW, AFW, and electrical power supply systems.			YES
C22	7450	PARTIAL LOSS OF CLASS IV AND CLASS III POWER	The event occurred during restarting from a planned outage (0.04 % full power, and the primary heat transport system depressurized with shutdown cooling as the in-service heat sink). An HE was made in re-aligning the electrical system (Class 4 – interruptible subsystem) after maintenance to 'normal' configuration (2 auxiliary transformers) that led to partial loss of Class 3 and 4 power buses. No reactor trip (IE)	PHWR	PWR(37)	
C23	7459	INCORRECT ALGORITHM IN MEMORY MODULE OF OVERLOAD PROTECTION DEVICE	The event occurred at power operation. To allow for maintenance work on a transformer a transfer of load to another supply route was conducted but was tripped by the associated protection device. This was due to an error in the software controlling this protection device. Incorrect software was installed also in other protection systems thus creating a mechanism for CCF. No IE	PWR		YES

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
C24	7468	MAIN FEEDWATER SYSTEM DEGRADATION IN SAFETY-RELATED ASME CODE CLASS 2 PIPING INSIDE THE CONTAINMENT OF A PRESSURIZED WATER REACTOR (NRC INFORMATION NOTICE 2001-09)	Generic report on the effects of erosion/corrosion on steel piping exposed to flowing water (single-phase fluids) and water-steam mixtures (two-phase fluids). Several cases of degradation in risk-important non-isolable sections of single-phase ASME Code Class 2 piping in MFW and main steam systems inside the containment were addressed. Failures of FW and other high-energy system components have resulted in complex challenges to operating staff when the released high-energy steam and water interacted with other systems, such as electrical distribution, fire protection, and security systems. It adversely affects the operability, availability, reliability, or function of systems required for safe shutdown and accident mitigation.	PWR	PWR(29)	YES
C25	7503	MULTI-UNIT TRANSIENT AND POTENTIAL LOSS OF HEAT SINK DUE TO BLOCKAGE OF COOLING WATER SCREENS BY ALGAE	The event involved 3 units that operated at full power. Two reactors were tripped (automatic tripping of condenser cooling water pumps on Unit 7 and Unit 8, followed by turbine trips and reactor setbacks on both units). Unit 6, the condenser cooling water pumps continued to run (the automatic trips disabled due to maintenance). The unit continued to operate throughout the event, but a large differential pressure across the traveling screens could have resulted in a loss of service water supply to all units on inadequate pump suction, or possibly the collapse of the traveling screens.			YES
C26	7563	UNIT POWER REDUCTION DUE TO TURBOGENERATOR TRIP BY THE EARTH PROTECTION SYSTEM AS A RESULT OF A PROBLEM WITH STATOR WINDING COOLING	The event occurred at power (rising power level to full power). One of the TGs was overheated and tripped on stator winding earth protection system, reactor power reduced to 50%. The cause was a manufacturing fault in the generator stator winding. No IE	LWGR	PWR(34)	
C27	7611	REACTOR REGULATION SYSTEM FAILED IN DIGITAL CONTROL COMPUTER Y, WITH DIGITAL CONTROL COMPUTER X SHUT DOWN FOR MAINTENANCE ACTIVITIES	The event occurred at low power (planned outage). The Digital Control Computer DCC-X was in maintenance and the second subsystem (DCC-Y) failed. Reactor shutdown manually. The cause was a foult in power supply.	PHWR	PWR(40)	
C28	7626	PLUGGING OF SAFETY INJECTION PUMP LUBRICATION OIL COOLERS WITH LAKEWEED: NRC INFORMATION NOTICE 2004-07	Several events with plugging and blockage of heat exchanger tubing (due to buildup of lakeweed or silt) observed at full power operation. This problem may decrease the ability of the heat exchanger or cooler to perform its required heat removal function. This may also affect other safety-related components, such as pumps; emergency diesel generators; and heating, ventilation, and air conditioning equipment, causing them to potentially fail when called upon to perform their safety-related function	PWR		YES
C29	7656	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS: NRC GENERIC LETTER 2004-02	Generic NRC report revising the guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS	PWR		YES
C30	7696	DROP IN FLOW RATE OF COOLING WATER TO THE TURBINE CONDENSER CAUSED BY FRAZIL ICE FORMATION IN THE COOLING POND AND CLOGGING OF PUMP STATION DEBRIS SCREENS	The event occurred at full power. Clogging of the debris screens with ice led to drop of the CCW flow, reduction of power and subsequently to turbine trip on condenser pressure signal (reactor power reduced to 5%)	PWR WWER		YES

#	IRS Number	Title	Description		IE type (EPRI)	In-depth analysis?
C31	7770	TRIPPING OF TWO REACTOR COOLANT PUMPS AS A RESULT OF A DROP IN OIL PRESSURE CAUSED BY THE OPENING OF A VALVE ON THE RECIRCULATION LINE FOLLOWING ACCIDENTAL ACTIVATION BY PERSONNEL OF THE ISOLATION VALVE END SWITCH	The event occurred at full power during decontamination work of the RCP oil system rooms (RCP-1 and RCP-3). Inadvertent activation of the limit switch for closing an isolation valve and trigerring actuation of design interlock for opening oil syste recirculation line that led to a drop of oil pressure and RCP trip. No IE	PWR WWER		
C32	7812	VIBRATION-INDUCED DEGRADATION AND FAILURE OF SAFETY-RELATED VALVES: NRC INFORMATION NOTICE 2006-15	Several failures were observed of the control valves in the AFW system. The valve degradation was attributed to the flow-induced metal fatigue failure of a cotter pin and blocking the flow through a restricting orifice by the pilot plug spacer and a washer. Degradation of safety-related system. No IE	PWR		YES
C33	7813	RECENT OPERATING EXPERIENCE OF SERVICE WATER SYSTEMS DUE TO EXTERNAL CONDITIONS: NRC INFORMATION NOTICE 2006-17	Generic report addressing multiple events that involved blockages in service water systems. The various blocking agents included silt, sand, small rocks, grass or weeds, frazil ice, and small aquatic fauna, such as fish. All these events were of low safety significance but illustrate the susceptibility of the safety-significant service water system. Degradation of essential support system			YES
C34	7817	NEW ULTRA-LOW-SULFUR DIESEL FUEL OIL COULD ADVERSELY IMPACT DIESEL ENGINE PERFORMANCE : NRC INFORMATION NOTICE 2006-22	Generic report addressing potential for new ultra-low-sulfur diesel (ULSD) fuel oil to adversely impact engine performance. This ULSD issue is of particular concern because it affects all licensee diesel generators that are safety-related and/or important to safety, thereby, presenting a possible common mode failure.			YES

* Indicate generic safety issues

7 Appendix B

Summary of CCI-related insights for the events selected for in-depth evaluation

#	REPORT #	EVENT REPORT TITLE	EVENT TYPE	EVENT DIRECT CAUSE	ADDITIONAL DEGRADED SYSTEMS	EVENT/ISSUE CATEGORY; DEPENDENCIES INVOLVED
1	1516	SUSCEPTIBILITY OF CONTAINMENT SUMP RECIRCULATION GATE VALVES TO PRESSURE LOCKING (NRC INFORMATION NOTICE 95-14)	IE: RCS LOCA (Note 1)	(Note 1)	ECCS, RHR	PDM, Pressure locking of sump gate valves under LOCA conditions,
2	1522	POTENTIAL FOR LOSS OF AUTOMATIC ENGINEERED SAFETY FEATURES ACTUATION (NRC INFORMATION NOTICE 95-10, AND INFORMATION NOTICE 95-10, SUPPLEMENT 1)	IE: Steam line break	(Note 1)	ESFAS	P CCI, Steam jet effect on non-safety power supply, induced electrical faults in safety systems,
3	1525	POTENTIAL PRESSURE-LOCKING OF SAFETY-RELATED POWER-OPERATED GATE VALVES (NRC INFORMATION 95-18)	IE: RCS LOCA (Note 1)	(Note 1)	ECCS, RHR	PDM, Pressure locking of sump gate valves under LOCA conditions,
4	1546	UNEXPECTED CLOGGING OF A RESIDUAL HEAT REMOVAL (RHR) PUMP STRAINER WHILE OPERATING IN SUPPRESSION POOL COOLING MODE (NRC BULLETIN 95-02)	IE: SRV inadvertently opened	Valve failure	RHR	CCF, Blockage of RHR pump strainers with sludge (iron oxide) and fibers (polymer);
5	1589	LEAK FROM RESIUDAL HEAT REMOVAL SYSTEM AT A LOCATION PRECLUDING ISOLATION	IE: RCS LOCA	Pipe break (vent nozzle)	RHR	CCI; RHR inoperable due to the break location,
6	1599	SUSCEPTIBILITY OF LOW-PRESSURE COOLANT INJECTION AND CORE SPRAY INJECTION VALVES TO PRESSURE LOCKING (NRC INFORMATION NOTICE 95-30)	IE: RCS LOCA (Note 1)	Potential valve failure	LP SI and CSI systems	PDM; Pressure locking of valves under LOCA conditions;
7	1601	INADEQUATE OFFSITE POWER SYSTEM VOLTAGES DURING DESIGN-BASIS EVENTS (NRC INFORMATION NOTICE 95-37)	IE: Main generator or turbine trip (Note 1)	(Note 1)	Essential EPS and ESF	PDM; Re-sequencing of the ESF loads onto the DGs by the plant site voltage regulation system,
8	1610	SLOW FIVE PERCENT SCRAM INSERTION TIMES CAUSED BY VITON DIAPHRAGMS IN SCRAM SOLENOID PILOT VALVES (NRC INFORMATION NOTICE 96-07)	IE: Reactor trip	Deficiency of scram solenoid pilot valves	RPS	CCF; Slower rate of CR insertion,
9	1622	MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT (NRC BULLETIN 96-02)	IE: Heavy load drop	Equipment failure or HE (Note 1)	Isolation condensers and the torus (BWR)	PDM; Non-isolable LOCA outside containment due to cask drop affecting also a safety equipment,
10	6392	DEENERGIZATION OF 6 kV ESSENTIAL BUS BY UNDUE ACTION OF THE EARTH FAULT PROTECTION AT KURSK NPP	IE: Partial loss of off-site power	Short-circuit in electrical equipment (Note 1)	MFW (minor degradation)	CCI; EPS panels de-energized due to spurious action of the earth-fault protection system of breakers,
11	7050	DEGRADATION OF COOLING WATER SYSTEMS DUE TO ICING (NRC INFORMATION NOTICE 96-36)	IE: Manual reactor trip	Degraded operability of relevant systems	CCW, SWS	CCI; Environmental icing conditions,
12	7052	UNEXPECTED OPENING OF MULTIPLE SAFETY RELIEF VALVES (NRC INFORMATION NOTICE 96-42)	IE: Manual reactor trip	Multiple opening of several SRVs		PDM; Transient in the DC control power to SRV logic module,

#	REPORT #	EVENT REPORT TITLE	EVENT TYPE	EVENT DIRECT CAUSE	ADDITIONAL DEGRADED SYSTEMS	EVENT/ISSUE CATEGORY; DEPENDENCIES INVOLVED
13	7057	NPP OPERATIONAL EVENTS RELATED TO ABNORMAL IMPACT OF ENVIRONMENT ON EQUIPMENT OPERATION IN THE COLD TIME OF THE YEAR	IE: Automatic scram or power reduction (Note 1)	Spurious I&C signals	SG, AFW, EFW, EPS	CCI; Freezing impulse lines related to secondary system parameters,
14	7105	IMPROPER ELECTRICAL GROUNDING RESULTS IN SIMULTANEOUS FIRES IN THE CONTROL ROOM AND THE SAFE-SHUTDOWN EQUIPMENT ROOM (NRC Information Notice 97-01)	Fire within plant (no reactor trip)	Fire due to an EPS design deficiency	ELUPS, fire detection and alarm system	P CCI; Fire in two separate plant locations due to a design error (grounding of isolation transformer),
15	7126	EXCESSIVE COOLDOWN AND DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM FOLLOWING LOSS OF OFFSITE POWER (NRC Information Notice 95-04, Supplement 1)	IE: Turbine/reactor trip	Short to ground of two EPS buses	Off-site power, RCS, charging system, seal injection system	CCI; Electrical short leading to loss of off-site power, a severe RCS overcooling transient, and transient induced failures of valves,
16	7129	IMPAIRMENT OF SHUTDOWN SYSTEM NO. 2 (SDS2) DUE TO MAINTENANCE ERROR - WRONG UNIT CALIBRATED	Potential IE	Human error	Reactivity control system	PDM; HE during calibration of SDS2 (ROP detectors),
17	7135	POTENTIAL NITROGEN ACCUMULATION RESULTING FROM BACKLEAKAGE FROM SAFETY INJECTION TANKS (NRC Information Notice 97-40)	Potential IE	Leaking isolation valves in SI test headers	ECCS, RHR	PDM; Gas voiding in LPI/RHR leading to waterhammer-induced pressure locking of valves,
18	7155	POTENTIAL FOR WATER HAMMER DURING RESTART OF RESIDUAL HEAT REMOVAL PUMPS (NRC Information Notice 87-10, Supplement 1)	IE: LOCA (Note 1)	Leaking safety relief valves	RHR	PDM; Re-alignment of RHR from SPC to LP SI mode and possibility of waterhammer damage,
19	7158	LEAKAGE FROM TURBINE BUILDING VENTILATION UNIT CAUSES UNIT WIDE LOSS OF ELECTRICAL POWER	IE: Loss of power to necessary plant systems	Electrical fault due to water dripping	EPS	P CCI; Loss of power supply to safe shutdown systems due to an area event (water dripping),
20	7163	REACTOR SCRAM DUE TO TYPHOON ATTACK	IE: Automatic reactor trip	External hazard induced faults	EPS, I&C air, MFW	CCI; High wind and EPS induced faults,
21	7168	RUPTURE IN EXTRACTION STEAM PIPING AS A RESULT OF FLOW-ACCELERATED CORROSION (NRC Information Notice 97-84)	IE: Manual reactor trip	Secondary system pipe break	Non-safety EPS	P CCI; Flow-accelerated corrosion and steam induced damage to EPS system,
22	7180	NUCLEAR POWER PLANT COLD WEATHER PROBLEMS AND PROTECTIVE MEASURES (NRC Information Notice 98-02)	Potential IE (Notes 1 and 2)	External hazard induced faults of equipment	CCW, SWS, EPS, ECCS automatics (note 2)	P CCI; Cold weather,
23	7192	WATER HAMMER EVENTS SINCE 1991 (NRC INFORMATION NOTICE 91-50, SUPPLEMENT 1)	Potential IE (Notes 1 and 2)	Water hammer	Secondary side systems (Note 2)	PDM; Water hammer,
24	7199	LOSS OF AUXILIARY POWER AT TWO UNITS OF BALAKOVO DUE TO BREAKER FAILURE IN THE 220KV OPEN SWITCHGEAR	IE: Reactor trip, LOSP	Deficiencies in EPS bus duct protection	Off-site power, EPS, failure of several motors in safety systems, Note 3)	P CCI; Humidity induced short circuit in the high voltage breaker and electrical transient leading to multiple faults affecting 2 units,

#	REPORT #	EVENT REPORT TITLE	EVENT TYPE	EVENT DIRECT CAUSE	ADDITIONAL DEGRADED SYSTEMS	EVENT/ISSUE CATEGORY; DEPENDENCIES INVOLVED
25	7214	POTENTIAL DEFICIENCY OF ELECTRICAL CABLE/CONNECTION SYSTEMS (NRC INFORMATION NOTICE 98-21)	Potential IE - LOCA (Note 1)	Thermal and radiation ageing of cables and/or connections	Various safety-related equipment (Note 1)	CCF; Potential steam-induced damage of cables and/or connections due to deficiencies in the environmental qualification,
26	7215	SHAFT BINDING IN GENERAL ELECTRIC TYPE SBM CONTROL SWITCHES (NRC INFORMATION NOTICE 98-19)	Several potential IEs, (Note 1)	Binding of control board switches	Various safety-related equipment (Note 1)	CCF; Material/design problem (post-mold cure shrinkage),
27	7228	MANUAL TRIP OF AUXILIARY FEEDWATER PUMPS DURING AUTOMATIC OPERATION	IE: Automatic reactor trip	Human errors	AFW	CCI; Cavitation and waterhammer,
28	7236	MANUAL REACTOR SCRAM DUE TO FAILURE OF SAFETY SYSTEM TRAIN 2 CAUSED BY ELECTRICAL EQUIPMENT FLOODING BY THE FIRE EXTINGUISHING SYSTEM WATER	IE: Manual trip	Area event (actuation of fire extinguishing system)	EPS, ESFAS	P CCI; Spurious initiation of fire distinguishing system due to a short circuit (deficiencies in fire detection system),
29	7241	REACTOR TRIP AND SUBSEQUENT POST-TRIP COOLING DEFICIENCIES	IE: Automatic reactor trip	Control relay coil failure in FW system	Post-trip cooling system	P CCI; Control relay operating at the extreme limits of its voltage tolerance,
30	7254	STEM BINDING IN TURBINE GOVERNOR VALVES IN REACTOR CORE ISOLATION COOLING (RCIC) AND AUXILIARY FEEDWATER (AFW) SYSTEMS (NRC INFORMATION NOTICE 98-24)	Potential IE (Note 1)	Design deficiency of turbine governor valve	RCIC, AFW	CCF; Stem binding of the RCIC turbine governor valve due to thermal expansion problems,
31	7257	MOTOR-OPERATED VALVE PERFORMANCE ISSUES (NRC INFORMATION NOTICE 96-48, SUPPLMENT 1)	Potential IE (note 1)	Design deficiency of MOVs	Various safety-related equipment (note 1)	CCF; Inappropriate sizing of AC-powered motor actuators for MOVs,
32	7264	LOSS OF CONSENSER VACUUM FOR JOINT CRACK PRODUCES TURBINE TRIP, REACTOR SCRAM, SIGNAL TO CLOSE MAIN STEAM ISOLATION VALVES FOR LOW CONDENSER VACUUM FAILS FOR WATER CONDENSATION ON CONDENSER PRESSURE SENSING LINES	IE: Automatic reactor trip (condenser vacuum loss)	Failure of steam pressure sensing lines	MSIVs, Post trip mitigation capabilities	P CCI; Condensation plugging of pressure sensing lines (design deficiencies),
33	7265	SPURIOUS OPENING OF PRESSURISER POWER OPERATED RELIEF VALVES RESULTING IN REACTOR SCRAM	IE: Reactor trip (PORVs leakage)	Failure of the PORV actuation circuitry	PORV control system	CCF; Failure of power supply module in the valve actuation circuitry,
34	7272	UNAVAILABILITY OF THE ECCS DUE TO RUPTURE OF A PIPE OF THE SG BLOWDOWN SYSTEM DURING THE ANNUAL OUTAGE OF UNIT 1	Need for draining of ECCS storage tank during annual refueling outage	Leakage of chemical treatment solution from SG to ECCS storage tank	ECCS	PDM; Insufficient capacity of the SG room drain system,
35	7281	AGEING OF CABLES IN THE STEAM GENERATOR SPACE OF LOVIISA REACTORS	Generic issue (Note 1)	Temperature induced cable ageing	Various safety-related equipment (note 1)	PDM; Deterioration of thermal insulation of the piping in the SG box and increase rate of ageing,

#	REPORT #	EVENT REPORT TITLE	EVENT TYPE	EVENT DIRECT CAUSE	ADDITIONAL DEGRADED SYSTEMS	EVENT/ISSUE CATEGORY; DEPENDENCIES INVOLVED
36	7292	DECLARATION OF SITE EMERGENCY AT HUNTERSTON B FOLLOWING TWO COMPLETE LOSSES OF ELECTRICAL GRID SUPPLY DURING A PERIOD OF BAD WEATHER.	IE: Loss of off-site power in 2 reactors	High wind impact	Essential EPS automatics	CCI; Degradation in automatic EPS reconfiguration features due to repeated events of loss of off-site power (during a several hours),
37	7303	ERRONEOUS SAFETY SYSTEM STATUS CONTROL AFTER OUTAGE	Generic issue (Note 1)	Maintenance management problems	Various safety-related equipment (Note 1)	PDM; Multiple problems due to complex work situation, time pressure, insufficient procedures,
38	7320	LUBRICATION-RELATED COMMON MODE FAILURES	Generic issue (Note 1)	Maintenance related error	Various safety-related equipment (Note 1)	CCF; Unsatisfactory use of lubricants in mechanical and electrical components,
39	7334	AUGMENTED INSPECTION TEAM - REACTOR TRIP WITH COMPLICATIONS AT INDIAN POINT 2 (INSPECTION REPORT NO. 50-247/99-08)	IE: Reactor trip and LOOP	Failures of breakers in EPS	AC and DC power, AFW	PDM; Failures in support system (EPS),
40	7339	RESOLUTION OF GENERIC ISSUE 145, ACTIONS TO REDUCE COMMON-CAUSE FAILURES (NRC REGULATORY ISSUE SUMMARY 99-03)	Generic issue (Note 1)	Note 1		CCF, Identified coupling factors include: maintenance practices, design problems, human errors
41	7342	PARTIAL LOSS OF SAFEGUARD SYSTEMS AS A RESULT OF EXTERNAL FLOODING	IE: Manual reactor trip	Flood-induced and wind-induced damage	SWS, ECCS, CCS, EPS (Off-site power)	CCI; Damage of equipment due to an external flooding (high tide) and loss of off-site power lines (high wind)
42	7344	ANGRA 1 REACTOR TRIP DUE TO LOSS OF THE PREFERRED OFF SITE POWER 138kV AND INVERTER BOP-2 FAILURE	IE: Automatic turbine trip & reactor trip	Failures of electrical equipment induced by grid disturbances	MFW, EPS (Off-site power)	P CCI, Disturbances in external grid leading to electrical transient and unavailability of multiple components
43	7352	DEGRADATION OF PIPE COMPONENTS ON RECIRCULATION SECTIONS OF CONTAINMENT SPRAY SYSTEM	IE: Manual shutdown for maintenance	CSS degradation identified during a test	CSS	CCF, Dependency due to inadequate design (modification) and badly assembled expansion joints
44	7365	LOSS OF POWER OF A 220 VAC SAFETY BUS BAR	IE: Manual reactor trip	EPS equipment failures	Turbine bypass, power for vital I&C	P CCI, Dependency through essential support system (electrical power supply)
45	7381	FAILURE OF COOLANT TEMPERATURE REGULATING VALVES OF EMERGENCY DIESEL GENERATORS	Generic issue identified during a DG power test	Failure of a valve in the DG cooling system	Vulnerability of EDG	CCF, design deficiency introduced as a result of modification
46	7404	AUTOMATIC TURBINE CONTROL SYSTEM DEGRADATION BECAUSE OF INCOMPATIBLE ELEMENTS USE DURING MAINTENANCE	Generic issue identified during a test	Failure of on-line memory module	Turbine automatic regulation/protectio n	CCF, Natural ageing of capacitors in the turbine control system (problem caused by insufficient experience feedback)
47	7410	NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER (NRC INFORMATION NOTICE 2000-14)	IE: Turbine trip	Electrical fault in an electrical bus duct	EPS, RCP, CCW	PDM; Electrical fault in a non-class bus duct and fire-induced failure of a vital EPS

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48	7411	LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION (NRC INFORMATION NOTICE 95-03, SUPPLEMENT 2)	Generic issue, LOCA in hot shutdown	Incorrect system alignment due to deficiency of a procedure	Potential degradation of ECCS	PDM, Incorrect procedure for alignment of the charging and SI systems under hot shutdown LOCA conditions (potential for multiple failures of pumps)
49	7436	EVALUATION OF AIR-OPERATED VALVES AT U.S. LIGHT-WATER REACTORS (NRC NUREG-1275, VOLUME 13)	Generic issue identified by testing (Note 1)	Failures of AOVs	Various safety systems	CCF, coupling mechanisms: design errors, contamination from the pneumatic system or from fabrication and maintenance activities.
50	7441	POTENTIAL LOSS OF REDUNDANT SAFETY- RELATED EQUIPMENT BECAUSE OF THE LACK OF HIGH-ENERGY LINE BREAK BARRIERS (NRC INFORMATION NOTICE 2000-20)	Generic issue (Note 1)	High energy line breaks	CCW, AFW, EPS	PDM; Multiple failures due to a harsh environment
51	7459	INCORRECT ALGORITHM IN MEMORY MODULE OF OVERLOAD PROTECTION DEVICE	Deficiency identified during maintenance	Failure of automatic switchover to alternative power supply	EPS (transformer)	CCF; Coupling factor: incorrect software controlling protection device installed in several systems
52	7468	MAIN FEEDWATER SYSTEM DEGRADATION IN SAFETY-RELATED ASME CODE CLASS 2 PIPING INSIDE THE CONTAINMENT OF A PRESSURIZED WATER REACTOR (NRC INFORMATION NOTICE 2001-09)	Generic issue	High energy line breaks	Various safety systems	PDM; Flow accelerated erosion/corrosion on steel piping of secondary systems and potential failure of other safety systems due to a harsh environment
53	7480	STRESS CORROSION CRACKING IN AUSTENITIC STAINLESS STEEL COMPONENTS	Generic issue identified during outage inspections	Environmental site-related conditions (high marine salt concentration)	ECCS, CSS	PDM; Degradation of SS piping walls due to a stress corrosion cracking
54	7495	LOSS OF SAFETY FUNCTION ON THE ENHANCED SHUTDOWN (ESD) SYSTEM DUE TO A WIRING DESIGN ERROR	Issue identified during routine maintenance	Unrevealed loss of reactor trip function	ESD	CCF; Loss of functionality of 20 ESD control rods due to a wiring design error
55	7502	DISCOVERY OF SHORTING LINKS WHICH HAD BEEN LEFT IN PLACE IN ERROR AND DEFEATED THE DOUBLE POLE/DOUBLE THROW TRIP SYSTEM ON THE FUELLING MACHINE	Deficiencies identified during testing	Potential for dropping irradiated fuel elements	Refueling machine	CCF; Shorting links (across two pairs of terminals in one of the control system cabinets may lead to unavailability of redundant channels (due to HE)
56	7503	MULTI-UNIT TRANSIENT AND POTENTIAL LOSS OF HEAT SINK DUE TO BLOCKAGE OF COOLING WATER SCREENS BY ALGAE	IE: Automatic reactor trip	Blockage of heat sink water intake	CCW (more than one unit affected)	PDM; Potential for loss of SWS system due to a blockage of water intake screens by algae
57	7530	SIMULTANEOUS LOSS OF TRAIN A CONTROBLOC AND TRAIN A ELECTRICAL SWITCHBOARDS FOR THE 6.6 KV AC NORMAL DISTRIBUTION SYSTEM AND 6.6 KV AC EMERGENCY SUPPLIED DISTRIBUTION SYSTEM	IE: Automatic reactor trip	HE during inverter testing	CVCS, AFW, CCW	CCI; Dependency mechanism: electrical power supply (loss of relevant switchboards, and control systems)

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58	7533	MANUAL SCRAM FOLLOWING LOSS OF 400 KV LINE	IE: Manual reactor trip	Electrical fault in off-site grid (HE)	MFW, CS	P CCI; Potential dependency through essential support system (electrical power lost due to electrical system transient caused by failure of automatic transfer to back-up power supply)
59	7534	CHOKING OF SCREENS AT CONFINEMENT SPRAY PUMP SUCTION	Deficiency identified during testing	Choking the screens at the containment sump	ECCS, CSS	PDM; Potential for unavailability of ECCS in response to LOCA due to a blockage of the containment sump screens (maintenance error)
60	7536	STEAM CONDENSATION-INDUCED WATER HAMMER IN A DUMP STEAM LINE, LEADING TO FAILURE OF REACTOR COOLING WATER PIPEWORK AND SHUTDOWN OF REACTOR 2	IE: Steam line break	Fracture of a pipe due to water hammer	Dump steam system	PDM; Dependency mechanism: condensation induced water hammer
61	7538	SHUTDOWN OF UNIT 2 DUE TO FIRE IN THE PRIMARY CIRCUIT CABLE SHAFT A110/2	IE: Manual shutdown	Fire in the cable room	Make-up system, water cleanup system, process parameter measuring	P CCI; Dependency mechanism: fire-induced damage to electrical systems
62	7547	MULTI-UNIT POWERHOUSE EQUIPMENT DAMAGE RESULTING FROM OPERATION OF EMERGENCY POWERHOUSE VENTING SYSTEM	Automatic load reduction	Generator hydrogen system failure	Instrument lines	PDM; Freezing of instrument lines when using the powerhouse emergency venting
63	7549	CRACKING OF A CONTROL ROD DRIVE MECHANISM PROTECTION PIPE	Problem identified during walk-down inspection	Corrosion damage	Temperature measurement device	CCF; Trans-granular stress corrosion cracking
64	7557	RECENT EXPERIENCE WITH REACTOR COOLANT SYSTEM LEAKAGE AND BORIC ACID CORROSION (NRC INFORMATION NOTICE 2003-02)	IE: RCS leakage	Corrosion damage	RCS barrier	CCF; Boric acid corrosion
65	7561	SAFETY INJECTION ACTUATED BY "VERY LOW PRESSURISER PRESSURE " PROTECTION CAUSED BY INAPPROPRIATE OPERATOR MANOEUVRE	IE: Inadvertent actuation of SI at low RCS pressure	Inappropriate operator manoeuvre	Potential for damage of RCS barrier	P CCI; Removing of interlock for SI actuation at low pressure (HE) led to a very severe pressure transient with the potential of inducing a damage to RCS integrity
66	7562	UNPLANNED CHANGES IN NUCLEAR POWER PLANT UNIT POWER DUE TO PERSONNEL ERRORS	IE: Automatic reactor trip	HE during maintenance of fire pump	RCP, CS, CC, SWS	P CCI; Power supply to several components interrupted by bus protection system
67	7569	PIPE RUPTURE INSIDE CONTAINMENT DUE TO RADIOLYSIS GASES EXPLOSION	IE: Manual load reduction	Explosion of radiolysis gases in RPVHS line	Potential damage to safety systems in the containment	PDM; Impact of explosion and leakage of steam inside the containment. Unforeseen interaction between systems
68	7570	NEUTRON FLUX OSCILLATIONS AFTER BYPASS OF HIGH PRESSURE FEEDWATER HEATERS	IE: Automatic reactor trip	Erroneous instrument signal in MFW control (bypass of the heater)	Potential for direct damage of fuel due to a severe transient	PDM; Significant or unforeseen interaction between systems leading to severe instability in the core power / flow map

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69	7583	INCREASED RISK TO KOEBERG NPP UNIT 2 AS A RESULT OF DEGRADED 6,6 KV BREAKERS AND FAILURE OF TWO DIESEL GENERATORS ON DEMAND	Increased number of failures of HV AC breakers	Design problem (inappropriate roller hardness)	Potential for impact on EPS & supported systems	CCF; Coupling factor: design weakness of the breakers
70	7600	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS - NRC BULLETIN 2003-01	IE: LOCA, HELB (Note 1)	Note 1	ECCS, CSS	P CCI; Potential for unavailability of ECCS in response to HELB due to a blockage of the containment sump screens by debris
71	7626	PLUGGING OF SAFETY INJECTION PUMP LUBRICATION OIL COOLERS WITH LAKEWEED: NRC INFORMATION NOTICE 2004-07	Problem identified during a routine inspection	Silt and biological blockage of ECCS HE	Potential for impact on all ECCS trains	CCF; Coupling factor: blockage of HE by lake weed and/or silt
72	7636	REACTOR SCRAM RESULTING FROM A LOSS OF NON-CLASS 1E UNINTERRUPTIBLE POWER SUPPLY AND ALERT CONDITION DECLARED AT LAGUNA VERDE NPP-1	IE: Manual reactor shutdown	Electrical fault	EPS, Recirculation flow control, FW control, monitoring	P CCI; Loss of power supply to multiple safety-related systems. Dependency through support (EPS) system
73	7656	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS: NRC GENERIC LETTER 2004-02	IE: LOCA, HELB (Note 1)	Note 1	ECCS, CSS	P CCI; Potential for unavailability of ECCS in response to LOCA/HELB due to a blockage of the containment sump screens by debris
74	7678	UNIT SCRAM ON "ONE OUT OF TWO OPERATING REACTOR COOLANT PUMPS TRIPPED" SIGNAL OWING TO REDUCED TURBINE-DRIVEN FEEDWATER PUMP PERFORMANCE CAUSED BY A FEEDWATER PIPE LEAK	IE: FW line break	Operation-induced erosion and corrosion	Measurement channels of the FW pump control system	P CCI; Potential impact on IE mitigation systems through a harsh environmental conditions
75	7687	SAFE SHUTDOWN OF KALPAKKAM-2 REACTOR FOLLOWING TSUNAMI STRIKE	IE: Loss of condenser cooling water	Flood induced by tsunami	CCW, SWS	CCI; Impact on an essential support system due to area event (external flood)
76	7696	DROP IN FLOW RATE OF COOLING WATER TO THE TURBINE CONDENSER CAUSED BY FRAZIL ICE FORMATION IN THE COOLING POND AND CLOGGING OF PUMP STATION DEBRIS SCREENS	IE: Turbine trip	Clogging of the debris screens in CCW intake	Potential for a loss of heat sink (SWS)	PDM; Clogging of water intake by ice potentially affecting IE mitigating system
77	7722	RISK OF CONTAINMENT SUMP SCREEN BLOCKAGE	IE: LOCA, (Note 1)	Note 1	ECCS, CSS	P CCI; Potential for unavailability of ECCS in response to LOCA due to a blockage of the containment sump screens by debris
78	7727	LOSS OF POWER OF A 220 VAC SAFETY BUS BAR	IE: Turbine trip and power reduction	Loss of AC bus voltage	Turbine bypass, plant monitoring	P CCI; Potential for unavailability of IE mitigation systems through support system (EPS)
79	7733	SAFE SHUTDOWN POTENTIALLY CHALLENGED BY UNANALYZED INTERNAL FLOODING EVENTS AND INADEQUATE DESIGN (ML051440302): LICENSEE EVENT REPORT 305-2005-004 (KEWAUNEE)	Degraded conditions found during inspection	Unsatisfactory protection of safety equipment against flood	Potential for unavailability of various equipment	P CCI; Potential impact on essential mitigation systems due to area event (flood)

#	REPORT #	EVENT REPORT TITLE	EVENT TYPE	EVENT DIRECT CAUSE	ADDITIONAL DEGRADED SYSTEMS	EVENT/ISSUE CATEGORY; DEPENDENCIES INVOLVED
80	7740	COMPLIANCE DEVIATION OF K1 CONNECTION BOXES	Degraded conditions	Leakage on the FW line	Potential for unavailability of various systems	P CCI; Potential impact on IE mitigation systems through environmental conditions in the containment (water spraying from the leak and faults in the electrical junction boxes).
81	7741	DEFECTIVE OPERATION OF 6.6 KV ELECTRICAL CIRCUIT BREAKERS	Increased rate of failures of 6.6 kV circuit breakers	Maintenance problem	Potential for impact on EPS & supported systems	CCF; Circuit breakers with little regular movement vulnerable to gumming effect due to grease ageing
82	7748	INSUFFICIENT WATER LEVEL IN SUMP 3 CSS 004 BA FOR RECIRCULATION ON CSS SPRAY LINE	Degraded conditions	Design/maintenance problem	Potential unavailability of ECCS and CSS	CCF; Unavailability of ECCS/CSS due to a ‘vapour lock’ effect (caused by migration of a pocket of air trapped in the ”check valve” pipe section)
83	7763	USE OF GALVANIZED SUPPORTS AND CABLE TRAYS WITH MEGGITT SI 2400 STAINLESS- STEEL-JACKETED ELECTRICAL CABLES: NRC INFORMATION NOTICE 2006-02	Degraded conditions	Design problem	Potential for unavailability of various systems	CCF; Coupling factor: vulnerability of stainless-steel-jacketed cables to degradation under fire conditions
84	7778	POSSIBLE DEFECT IN BUSSMANN KWN-R AND KTN-R FUSES, NRC INFORMATION NOTICE 2006-05	Degraded conditions	Poor solder joints in electrical fuses	Potential for unavailability of various systems	CCF; Coupling factor: fabrication fault
85	7781	MANUAL REACTOR TRIP DUE TO LOSS OF ELECTRICAL DISTRIBUTION BOARD	IE: Manual reactor trip	Plant shutdown to investigate and repair 3 failed inverters	EPS (AC control and instrumentation)	P CCI; Potential impact on IE mitigation systems through support system (dry joint on a soldered connection on the voltage divider printed circuit board causing multiple failures of EPS components)
86	7788	LOSS OF 400 KV AND SUBSEQUENT FAILURE TO START EMERGENCY DIESEL GENERATORS IN SUB A AND SUB B	IE: Automatic reactor trip	Disturbances in the off-site 400 kV switchyard and a weakness of the design	EPS, EDG automatics, MCR indications	CCI; IE impact on mitigation systems through essential support (EPS) system, selectivity of the bus bar protections not adequate to protect the inverters
87	7795	SAFETY INJECTION AND REACTOR TRIP DUE TO AN UNSKILLFUL OPERATION DURING PLANT STARTUP	IE: Opening of the MS SV and RV (PORV)	Transient in the MS system due to operator error	Potential for damage to ECCS, CSS due to a flying missile	P CCI; Potential impact on IE mitigation systems through the damage by transient-induced missile (pipe segment silencer of the PORV)
88	7801	ANGRA 1 TRIP DUE TO TOTAL LOSS OF OFFSITE POWER (LOPS) AND AUXILIARY FEEDWATER TURBINE-DRIVEN PUMP TRIP	IE: Loss of offsite power	External grid problem	AFW	CCI; IE mitigation system (AFW) tripped due to a load rejection transient (pump over-speed protection)

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89	7812	VIBRATION-INDUCED DEGRADATION AND FAILURE OF SAFETY-RELATED VALVES: NRC INFORMATION NOTICE 2006-15	Increased rate of failures of the AFW control valves	Valve design problem and environmental conditions	AFW	CCF; Coupling factor: impact of vibration (flow-induced metal fatigue failure of a cotter pin)
90	7813	RECENT OPERATING EXPERIENCE OF SERVICE WATER SYSTEMS DUE TO EXTERNAL CONDITIONS: NRC INFORMATION NOTICE 2006-17	Generic problem	Blocking of flow by foreign objects	SWS	CCF; Coupling factor: blocking of flow by silt, sand, small rocks, grass or weeds, frazil ice, and small aquatic fauna.
91	7817	NEW ULTRA-LOW-SULFUR DIESEL FUEL OIL COULD ADVERSELY IMPACT DIESEL ENGINE PERFORMANCE : NRC INFORMATION NOTICE 2006-22	Generic problem	Use of new ultra-low-sulfur diesel (ULSD) fuel oil	EDG	CCF; Coupling factor: Use of new fuel oil that adversely affects DG engine performance

Notes to Table:

1) Generic report that is not associated with a specific event/condition (e.g. indicates a potential degradation problem) and/or addresses several events/conditions of similar type; the entry regarding IE in the column 'Event Type' describes an applicable (potential) IE
2) IRS generic report refers to several events/plants; information in the column 'Additional Degraded Systems' includes a summary of all events/plants
3) Information in the IRS report regarding the induced failures and affected systems not sufficient.

8 Appendix C

Input for the categorization of the events selected for in-depth evaluation

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
1516	SUSCEPTIBILITY OF CONTAINMENT SUMP RECIRCULATION GATE VALVES TO PRESSURE LOCKING (NRC INFORMATION NOTICE 95-14)	PDM	-	Pipe break (Notes 1 & 2)	M/M	Environment/ Area	Pressure impact on valves/ accident-induced
1522	POTENTIAL FOR LOSS OF AUTOMATIC ENGINEERED SAFETY FEATURES ACTUATION (NRC INFORMATION NOTICE 95-10, AND INFORMATION NOTICE 95-10, SUPPLEMENT 1)	P CCI	-	Pipe break	M/M	Environment/ Area	Steam jet impact on electrical equipment
1525	POTENTIAL PRESSURE-LOCKING OF SAFETY-RELATED POWER-OPERATED GATE VALVES (NRC INFORMATION 95-18)	PDM	-	Pipe break (Notes 1 & 2)	M/M	Environment/ Area	Pressure impact on valves/ accident-induced
1546	UNEXPECTED CLOGGING OF A RESIDUAL HEAT REMOVAL (RHR) PUMP STRAINER WHILE OPERATING IN SUPPRESSION POOL COOLING MODE (NRC BULLETIN 95-02)	CCF	-	Valve inadvertently opened	M/M	Common Cause	Flow blockage/ internal agent (sludge, fibbers)/ strainers in the redundant trains
1589	LEAK FROM RESIUDAL HEAT REMOVAL SYSTEM AT A LOCATION PRECLUDING ISOLATION	CCI	-	Pipe break	M/M	Direct/ physical	Break impact on safety system (loss of flow through the break)
1599	SUSCEPTIBILITY OF LOW-PRESSURE COOLANT INJECTION AND CORE SPRAY INJECTION VALVES TO PRESSURE LOCKING (NRC INFORMATION NOTICE 95-30)	PDM	-	Pipe break (Notes 1 & 2)	M/M	Environment/ Area	Pressure impact on valves/ accident induced
1601	INADEQUATE OFFSITE POWER SYSTEM VOLTAGES DURING DESIGN-BASIS EVENTS (NRC INFORMATION NOTICE 95-37)	PDM	-	Turbine/generator trip	M/E	Support	Undesirable action of control system on safety system
1610	SLOW FIVE PERCENT SCRAM INSERTION TIMES CAUSED BY VITON DIAPHRAGMS IN SCRAM SOLENOID PILOT VALVES (NRC INFORMATION NOTICE 96-07)	CCF	-	Valve failure	M/M	Common Cause	Design deficiency/ material problem
1622	MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT (NRC BULLETIN 96-02)	PDM	OE+ (Note 1)	Cask handling fault (Notes 1 & 2)	M/M	External	Heavy load drop

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
6392	DEENERGIZATION OF 6 kV ESSENTIAL BUS BY UNDUE ACTION OF THE EARTH FAULT PROTECTION AT KURSK NPP	CCI	-	Failure of electric motor	E/E	Support	Spurious action of equipment protection
7050	DEGRADATION OF COOLING WATER SYSTEMS DUE TO ICING (NRC INFORMATION NOTICE 96-36)	CCI	-	Water intake screens blocked by ice	M/M	Environment/ Area	Cold weather
7052	UNEXPECTED OPENING OF MULTIPLE SAFETY RELIEF VALVES (NRC INFORMATION NOTICE 96-42)	PDM	-	Valves inadvertent opening	I/I	Protection/ Control	Transient/ electrical/ control system logic module
7057	NPP OPERATIONAL EVENTS RELATED TO ABNORMAL IMPACT OF ENVIRONMENT ON EQUIPMENT OPERATION IN THE COLD TIME OF THE YEAR	CCI	ME+	Impulse lines frozen	M/I	Environment/ Area	Cold weather
7105	IMPROPER ELECTRICAL GROUNDING RESULTS IN SIMULTANEOUS FIRES IN THE CONTROL ROOM AND THE SAFE-SHUTDOWN EQUIPMENT ROOM (NRC Information Notice 97-01)	P CCI	DE+	Short circuit due to improper electrical grounding of equipment	E/E	Environment/ Area	Fire/electrical short-induced
7126	EXCESSIVE COOLDOWN AND DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM FOLLOWING LOSS OF OFFSITE POWER (NRC Information Notice 95-04, Supplement 1)	CCI	-	Bus short to ground	E/E	Transient	Transient/ electrical/ internal cause & thermal transient
7129	IMPAIRMENT OF SHUTDOWN SYSTEM NO. 2 (SDS2) DUE TO MAINTENANCE ERROR - WRONG UNIT CALIBRATED	PDM	ME*	Calibration action on a wrong unit	I/I	Maintenance	Maintenance error/ calibration
7135	POTENTIAL NITROGEN ACCUMULATION RESULTING FROM BACKLEAKAGE FROM SAFETY INJECTION TANKS (NRC Information Notice 97-40)	PDM	-	Leakage of Nitrogen-saturated water (Notes 1 & 2)	M/M	Transient	Water hammer
7155	POTENTIAL FOR WATER HAMMER DURING RESTART OF RESIDUAL HEAT REMOVAL PUMPS (NRC Information Notice 87-10, Supplement 1)	PDM	OE+	SRV leakage + LOCA (Notes 1 & 2)	M/M	Transient	Water hammer
7158	LEAKAGE FROM TURBINE BUILDING VENTILATION UNIT CAUSES UNIT WIDE LOSS OF ELECTRICAL POWER	P CCI	-	Leakage from a ventilation unit	M/M	Environment/ Area	Water spray
7163	REACTOR SCRAM DUE TO TYPHOON ATTACK	CCI	-	Breaker trip by external cause	E/E	External	High wind
7168	RUPTURE IN EXTRACTION STEAM PIPING AS A RESULT OF FLOW-ACCELERATED CORROSION (NRC Information Notice 97-84)	P CCI	-	Steam line rupture due to flow-accelerated corrosion	M/M	Environment/ Area	Harsh environment/ steam/ jet impact

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
7180	NUCLEAR POWER PLANT COLD WEATHER PROBLEMS AND PROTECTIVE MEASURES (NRC Information Notice 98-02)	P CCI	-	(Note 3) Various causes including ice plug in valve or pipe, impulse line freezing	M/M M/I	Environment/ Area	Cold weather
7192	WATER HAMMER EVENTS SINCE 1991 (NRC INFORMATION NOTICE 91-50, SUPPLEMENT 1)	PDM	ME+ DE+	Void collapse/condensation-induced pressure transients (Note 3)	M/M	Transient	Water hammer
7199	LOSS OF AUXILIARY POWER AT TWO UNITS OF BALAKOVO DUE TO BREAKER FAILURE IN THE 220KV OPEN SWITCHGEAR	P CCI	-	Electrical breaker short circuit	E/E	Environment/ Area	Harsh environment/ humidity
7214	POTENTIAL DEFICIENCY OF ELECTRICAL CABLE/CONNECTION SYSTEMS (NRC INFORMATION NOTICE 98-21)	CCF	-	Potential failure of cables due to ageing (Note 1)	E/E	Environment/ Area	Harsh environment/ steam/ impact on cables
7215	SHAFT BINDING IN GENERAL ELECTRIC TYPE SBM CONTROL SWITCHES (NRC INFORMATION NOTICE 98-19)	CCF	-	Control board switch binding (Note 1)	I/I	Common Cause	Design/material problem
7228	MANUAL TRIP OF AUXILIARY FEEDWATER PUMPS DURING AUTOMATIC OPERATION	CCI	OE*	Pressure transient due to HE, pump seal leakage	M/M	Transient	Water hammer/ cavitation
7236	MANUAL REACTOR SCRAM DUE TO FAILURE OF SAFETY SYSTEM TRAIN 2 CAUSED BY ELECTRICAL EQUIPMENT FLOODING BY THE FIRE EXTINGUISHING SYSTEM WATER	P CCI	DE+	Spurious signal in the fire extinguishing system	I/I	Environment/ Area	Water spray
7241	REACTOR TRIP AND SUBSEQUENT POST-TRIP COOLING DEFICIENCIES	P CCI	OE+	Control relay failure	I/I	Protection/ Control	Operational deficiency/ setpoints
7254	STEM BINDING IN TURBINE GOVERNOR VALVES IN REACTOR CORE ISOLATION COOLING (RCIC) AND AUXILIARY FEEDWATER (AFW) SYSTEMS (NRC INFORMATION NOTICE 98-24)	CCF	DE+	Turbine governor valve failure (stem binding)	M/M	Common Cause	Binding - mechanical/ design problem / thermal expansion
7257	MOTOR-OPERATED VALVE PERFORMANCE ISSUES (NRC INFORMATION NOTICE 96-48, SUPPLMENT 1)	CCF	DE+	Potential failures of MOVs due to inappropriate motor actuator sizing (Note 1)	E/M	Common Cause	Design deficiency /motor actuator sizing
7264	LOSS OF CONSENSER VACUUM FOR JOINT CRACK PRODUCES TURBINE TRIP, REACTOR SCRAM, SIGNAL TO CLOSE MAIN STEAM ISOLATION VALVES FOR LOW CONDENSER VACUUM FAILS FOR WATER CONDENSATION ON CONDENSER PRESSURE SENSING LINES	P CCI	-	Crack in the turbine exhaust pipe	M/M	Protection/ Control	Flow blockage/ condensation plugging of instrument lines

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
7265	SPURIOUS OPENING OF PRESSURISER POWER OPERATED RELIEF VALVES RESULTING IN REACTOR SCRAM	CCF	-	Spurious operation of PORVs	E/I	Support	Power supply module to redundant equipment
7272	UNAVAILABILITY OF THE ECCS DUE TO RUPTURE OF A PIPE OF THE SG BLOWDOWN SYSTEM DURING THE ANNUAL OUTAGE OF UNIT 1	PDM	DE+	Leakage in the SG blow-down system	M/M	Environment/ Area	Flood/ internal cause
7281	AGEING OF CABLES IN THE STEAM GENERATOR SPACE OF LOVIISA REACTORS	PDM	-	Potential failures of electrical cable due to ageing	E/E	Environment / Area	Temperature-accelerated ageing of cables
7292	DECLARATION OF SITE EMERGENCY AT HUNTERSTON B FOLLOWING TWO COMPLETE LOSSES OF ELECTRICAL GRID SUPPLY DURING A PERIOD OF BAD WEATHER.	CCI	-	Loss of off-site grid due to high wind	E/E	External	High wind (repeated events)
7303	ERRONEOUS SAFETY SYSTEM STATUS CONTROL AFTER OUTAGE	PDM	ME*	Equipment erroneously left inoperable due to human errors – 9 events (Note 3)	M/M E/E	Maintenance	Maintenance / management (complex work conditions)
7320	LUBRICATION-RELATED COMMON MODE FAILURES	CCF	ME*	Failure of mechanical and electrical components (Note 3)	M/M E/E	Maintenance	Maintenance/ Inappropriate lubricants
7334	AUGMENTED INSPECTION TEAM - REACTOR TRIP WITH COMPLICATIONS AT INDIAN POINT 2 (INSPECTION REPORT NO. 50-247/99-08)	PDM		Circuit breaker failures	E/E	Support	Power supply
7339	RESOLUTION OF GENERIC ISSUE 145, ACTIONS TO REDUCE COMMON-CAUSE FAILURES (NRC REGULATORY ISSUE SUMMARY 99-03)	CCF	ME* DE*	Potential for common cause failures of equipment (Note 1)	M/M E/E	Maintenance	Maintenance, design, human
7342	PARTIAL LOSS OF SAFEGUARD SYSTEMS AS A RESULT OF EXTERNAL FLOODING	CCI	-	Flood/wind-induced failures of equipment	M/M E/E	External	External flooding and high wind
7344	ANGRA 1 REACTOR TRIP DUE TO LOSS OF THE PREFERRED OFF SITE POWER 138kV AND INVERTER BOP-2 FAILURE	P CCI	-	Grid disturbance-induced failures of electrical components	E/E	Transient	Transient/ electrical/ grid disturbances
7352	DEGRADATION OF PIPE COMPONENTS ON RECIRCULATION SECTIONS OF CONTAINMENT SPRAY SYSTEM	CCF	DE+ OE+	Failure of expansion bellows	M/M	Common Cause	Design deficiency / pressure loading of redundant trains during a test

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
7365	LOSS OF POWER OF A 220 VAC SAFETY BUS BAR	P CCI	ME+	Inverter failure	E/E	Support	Power supply
7381	FAILURE OF COOLANT TEMPERATURE REGULATING VALVES OF EMERGENCY DIESEL GENERATORS	CCF	-	Malfunctions of control valves in EDG cooling system	M/M	Common Cause	Vibration induced fatigue/ design modification
7404	AUTOMATIC TURBINE CONTROL SYSTEM DEGRADATION BECAUSE OF INCOMPATIBLE ELEMENTS USE DURING MAINTENANCE	CCF		Failure of on-line memory module	I/I	Protection/ Area	Ageing/ electrical capacitors
7410	NON-VITAL BUS FAULT LEADS TO FIRE AND LOSS OF OFFSITE POWER (NRC INFORMATION NOTICE 2000-14)	PDM		Short circuit in the electrical bus duct	E/E	Environment/ Area	Fire/ induced by electrical fault
7411	LOSS OF REACTOR COOLANT INVENTORY AND POTENTIAL LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION (NRC INFORMATION NOTICE 95-03, SUPPLEMENT 2)	PDM	OE+	Potential for ECCS failure under LOCA due to incorrect procedure for system alignment (Note 1)	M/M	Common Cause	Operation/ procedure/ system alignment
7436	EVALUATION OF AIR-OPERATED VALVES AT U.S. LIGHT-WATER REACTORS (NRC NUREG-1275, VOLUME 13)	CCF	DE+	Potential for failure of AOVs (Note 1)	M/M	Common Cause	Binding – mechanical/ design errors
7441	POTENTIAL LOSS OF REDUNDANT SAFETY- RELATED EQUIPMENT BECAUSE OF THE LACK OF HIGH-ENERGY LINE BREAK BARRIERS (NRC INFORMATION NOTICE 2000-20)	PDM	-	Potential for high energy line break (Notes 1&2)	M/M	Environment/ Area	Harsh environment/ steam
7459	INCORRECT ALGORITHM IN MEMORY MODULE OF OVERLOAD PROTECTION DEVICE	CCF	DE+	Malfunction of the electrical protection device (load transfer automatics)	I/E	Support	Protection device / software fault
7468	MAIN FEEDWATER SYSTEM DEGRADATION IN SAFETY-RELATED ASME CODE CLASS 2 PIPING INSIDE THE CONTAINMENT OF A PRESSURIZED WATER REACTOR (NRC INFORMATION NOTICE 2001-09)	PDM	-	Potential for high energy line break (Notes 1&2)	M/M	Common Cause	Corrosion damage to barriers/ flow accelerated corrosion
7480	STRESS CORROSION CRACKING IN AUSTENITIC STAINLESS STEEL COMPONENTS	PDM	-	Potential pipe breaks (Notes 1&2)	M/M	Common Cause	Corrosion/ stress-induced cracking/ damage to barriers
7495	LOSS OF SAFETY FUNCTION ON THE ENHANCED SHUTDOWN (ESD) SYSTEM DUE TO A WIRING DESIGN ERROR	CCF	DE+	Malfunction of CR drives	I/M	Common Cause	Design – wiring problem/ common fault in redundant components

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
7502	DISCOVERY OF SHORTING LINKS WHICH HAD BEEN LEFT IN PLACE IN ERROR AND DEFEATED THE DOUBLE POLE/DOUBLE THROW TRIP SYSTEM ON THE FUELLING MACHINE	CCF	DE+	Short links in an electrical cabinet (protection of fuelling machine)	I/I	Common Cause	Maintenance/ shorting links not removed after testing
7503	MULTI-UNIT TRANSIENT AND POTENTIAL LOSS OF HEAT SINK DUE TO BLOCKAGE OF COOLING WATER SCREENS BY ALGAE	PDM		Blockage of water intake	M/M	External	Flow blockage/ bio-fouling
7530	SIMULTANEOUS LOSS OF TRAIN A CONTROBLOC AND TRAIN A ELECTRICAL SWITCHBOARDS FOR THE 6.6 KV AC NORMAL DISTRIBUTION SYSTEM AND 6.6 KV AC EMERGENCY SUPPLIED DISTRIBUTION SYSTEM	CCI	TE*	Loss of switchboard	E/E	Support	Power supply
7533	MANUAL SCRAM FOLLOWING LOSS OF 400 KV LINE	P CCI	OE*	Short circuit in the off-site distribution station	E/E	Support	Power supply
7534	CHOKING OF SCREENS AT CONFINEMENT SPRAY PUMP SUCTION	PDM	ME*	Spray pump intake screens blocked by impurities (detected during a test)	M/M	Environment/ Area	Potential flow blockage/ mechanical impurities in ECCS tanks
7536	STEAM CONDENSATION-INDUCED WATER HAMMER IN A DUMP STEAM LINE, LEADING TO FAILURE OF REACTOR COOLING WATER PIPEWORK AND SHUTDOWN OF REACTOR 2	PDM	-	Fracture of a pipe	M/M	Transient	Water hammer/ condensation induced
7538	SHUTDOWN OF UNIT 2 DUE TO FIRE IN THE PRIMARY CIRCUIT CABLE SHAFT A110/2	P CCI	-	False signals and RCP trip due to cable room fire	I/I E/I	Environment/ Area	Fire/ cable shaft
7547	MULTI-UNIT POWERHOUSE EQUIPMENT DAMAGE RESULTING FROM OPERATION OF EMERGENCY POWERHOUSE VENTING SYSTEM	PDM	-	Leakage from the generator hydrogen system	M/M	Environment/ Area	Cold weather and the use of emergency ventilation
7549	CRACKING OF A CONTROL ROD DRIVE MECHANISM PROTECTION PIPE	CCF	-	Cracking of CRD protection pipe	M/M	Common Cause	Corrosion damage to barriers/ stress corrosion cracking
7557	RECENT EXPERIENCE WITH REACTOR COOLANT SYSTEM LEAKAGE AND BORIC ACID CORROSION (NRC INFORMATION NOTICE 2003-02)	CCF	-	Corrosion cracking of RCS barrier	M/M	Common Cause	Corrosion damage to barriers/ Boron acid corrosion
7561	SAFETY INJECTION ACTUATED BY "VERY LOW PRESSURISER PRESSURE " PROTECTION CAUSED BY INAPPROPRIATE OPERATOR MANOEUVRE	P CCI	OE*	Incorrect operator action leading to a severe pressure transient	I/M	Transient	Pressure transient / lack of proper interlock

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
7562	UNPLANNED CHANGES IN NUCLEAR POWER PLANT UNIT POWER DUE TO PERSONNEL ERRORS	P CCI	ME*	Incorrect maintenance action leading to a loss of power supply	I/M	Support	Power supply / undesirable action of protection device
7569	PIPE RUPTURE INSIDE CONTAINMENT DUE TO RADIOLYSIS GASES EXPLOSION	PDM	-	Pipe break due to explosion of radiolysis gases	M/M	Environment/ Area	Explosion of radiolysis gas, harsh steam environment
7570	NEUTRON FLUX OSCILLATIONS AFTER BYPASS OF HIGH PRESSURE FEEDWATER HEATERS	PDM	-	Erroneous instrument signal	I/I	Transient	Severe thermal transient/ due to core power instability
7583	INCREASED RISK TO KOEBERG NPP UNIT 2 AS A RESULT OF DEGRADED 6,6 KV BREAKERS AND FAILURE OF TWO DIESEL GENERATORS ON DEMAND	CCF	DE+	Potential for common cause failure of circuit breakers (Note 1)	E/E	Common cause	Design deficiency/ material
7600	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS - NRC BULLETIN 2003-01	P CCI	-	Pipe break of HELB type (Notes 1 and 2)	M/M	Environment/ Area	Flow blockage / screens blocked by debris
7626	PLUGGING OF SAFETY INJECTION PUMP LUBRICATION OIL COOLERS WITH LAKEWEED: NRC INFORMATION NOTICE 2004-07	CCF	-	Heat exchanger tube blockage	M/M	External	Flow blockage/ bio-fouling
7636	REACTOR SCRAM RESULTING FROM A LOSS OF NON-CLASS 1E UNINTERRUPTIBLE POWER SUPPLY AND ALERT CONDITION DECLARED AT LAGUNA VERDE NPP-1	P CCI	-	Failure of a relay in the rectifier control module	I/I	Support	Power supply
7656	POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS: NRC GENERIC LETTER 2004-02	P CCI	-	Pipe break of HELB type (Notes 1 and 2)	M/M	Environment/ Area	Flow blockage / screens blocked by debris
7678	UNIT SCRAM ON "ONE OUT OF TWO OPERATING REACTOR COOLANT PUMPS TRIPPED" SIGNAL OWING TO REDUCED TURBINE-DRIVEN FEEDWATER PUMP PERFORMANCE CAUSED BY A FEEDWATER PIPE LEAK	P CCI	-	Break of pipe (FW bypass line)	M/M	Environment/ Area	Harsh environment/ steam
7687	SAFE SHUTDOWN OF KALPAKKAM-2 REACTOR FOLLOWING TSUNAMI STRIKE	CCI	-	Pump trip due to a flood	I/I E/E	External	External flood

REPORT	EVENT TITLE	EVENT TYPE	ELEMENTS INVOLVED IN A DIRECT CAUSE			DEPENDENCY MECHANISM	
			Human Action	Failure/fault description	SSC type	Type	Description
7696	DROP IN FLOW RATE OF COOLING WATER TO THE TURBINE CONDENSER CAUSED BY FRAZIL ICE FORMATION IN THE COOLING POND AND CLOGGING OF PUMP STATION DEBRIS SCREENS	PDM	-	CCW intake clogging by ice	M/M	Environment/ Area	Cold weather /ice
7722	RISK OF CONTAINMENT SUMP SCREEN BLOCKAGE	P CCI	-	Pipe break of HELB type (Notes 1 and 2)	M/M	Environment/ Area	Flow blockage / screens blocked by debris
7727	LOSS OF POWER OF A 220 VAC SAFETY BUS BAR	P CCI	-	Inverter failure	E/E	Support	Power supply
7733	SAFE SHUTDOWN POTENTIALLY CHALLENGED BY UNANALYZED INTERNAL FLOODING EVENTS AND INADEQUATE DESIGN (ML051440302): LICENSEE EVENT REPORT 305-2005-004 (KEWAUNEE)	P CCI	DE+	Internal flooding events + deficiencies in equipment protection (Notes 1 and 2)	M/M	Environment/ Area	Internal flood
7740	COMPLIANCE DEVIATION OF K1 CONNECTION BOXES	P CCI	DE+ ME+	Pipe break/leakage + deficiencies in electrical junction boxes (Notes 1 and 2	M/M	Environment/ Area	Water spray
7741	DEFECTIVE OPERATION OF 6.6 KV ELECTRICAL CIRCUIT BREAKERS	CCF	DE+ ME+	Failure of circuit breakers due to a gumming effect (Note 1)	E/E	Common Cause	Maintenance/ grease ageing
7748	INSUFFICIENT WATER LEVEL IN SUMP 3 CSS 004 BA FOR RECIRCULATION ON CSS SPRAY LINE	CCF	ME+	Valve failure due to vapour lock effect	M/M	Common Cause	Air penetration/ Vapour lock effect
7763	USE OF GALVANIZED SUPPORTS AND CABLE TRAYS WITH MEGGITT SI 2400 STAINLESS- STEEL-JACKETED ELECTRICAL CABLES: NRC INFORMATION NOTICE 2006-02	CCF	DE+	Fire + deficiency of electrical cables (Notes 1 and 2)	E/I	Common Cause	Design deficiency / cables/ susceptibility to fire
7778	POSSIBLE DEFECT IN BUSSMANN KWN-R AND KTN-R FUSES, NRC INFORMATION NOTICE 2006-05	CCF	DE+	Failure of electrical fuses	E/E	Common Cause	Component fabrication fault/ solder joints
7781	MANUAL REACTOR TRIP DUE TO LOSS OF ELECTRICAL DISTRIBUTION BOARD	P CCI	-	Failure of I&C card (loose connection)	I/E	Support	Power supply / fabrication/ faulty soldered connections
7788	LOSS OF 400 KV AND SUBSEQUENT FAILURE TO START EMERGENCY DIESEL GENERATORS IN SUB A AND SUB B	CCI	DE+	Loss of bus due to deficiency of the bus bar protection	E/E	Support	Equipment protection (inadequate bus bar protection)

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			Human Action	Failure/fault description	SSC type	Type	Description
7795	SAFETY INJECTION AND REACTOR TRIP DUE TO AN UNSKILLFUL OPERATION DURING PLANT STARTUP	P CCI	OE+	Opening of safety valve due to an operator error	M/M	Transient	Transient-induced missile
7801	ANGRA 1 TRIP DUE TO TOTAL LOSS OF OFFSITE POWER (LOPS) AND AUXILIARY FEEDWATER TURBINE-DRIVEN PUMP TRIP	CCI	ME+	Loss of off-site power + deficiency in pump protection settings	E/E I/M	Transient	Load rejection transient (impact on pump protection)
7812	VIBRATION-INDUCED DEGRADATION AND FAILURE OF SAFETY-RELATED VALVES: NRC INFORMATION NOTICE 2006-15	CCF	-	Failure of control valves due to flow-induced fatigue	M/M	Common Cause	Vibration induced fatigue
7813	RECENT OPERATING EXPERIENCE OF SERVICE WATER SYSTEMS DUE TO EXTERNAL CONDITIONS: NRC INFORMATION NOTICE 2006-17	CCF	-	Flow paths blockage	M/M	External	Flow blockage (fouling by foreign material)
7817	NEW ULTRA-LOW-SULFUR DIESEL FUEL OIL COULD ADVERSELY IMPACT DIESEL ENGINE PERFORMANCE : NRC INFORMATION NOTICE 2006-22	CCF	-	Decreased performance of DGs due to fuel features	M/E	Common Cause	Operational problem/ fuel adverse impact on DG performance

Notes:

[1]) IRS report addresses a potential degradation of equipment/system or a specific susceptibility of the plant, not a 'real' initiating event.

[2]) Information provided in the table on "elements involved in a direct cause" refers to a 'potential occurrence' (or IE) that would lead to a sequence of dependent occurrences (described in the columns "Dependency mechanism")

[3]) IRS generic report that refers to several events of similar type (often with slightly different direct causes)

Abbreviations used in Table

Human actions		Failures/faults	
TA	testing activities	M	Mechanical (component / system)
MA	maintenance activities	E	Electrical (component / system)
TE	testing error	I	Instrumentation & control (component / system)
ME	maintenance error		
OE	operational error		
DE	design error		
*	human error / no hardware failures involved		
+	human error / additional hardware failures involved		

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