SKI Report 98:09

Identification of Common Cause Initiators in IRS Database

Ralph Nyman
Maciej Kulig
Bojan Tomic

Februari 1998

ISSN 1104-1374
ISRN SKI-R-98/09--SE

SKI Report 98:09

Identification of Common Cause Initiators in IRS Database

Ralph Nyman [1]
Maciej Kulig [2]
Bojan Tomic [2]

[1] **Swedish Nuclear Power Inspectorate**
Department of Plant Safety Assessment (SKI/RA)

[2] **ENCONET Consulting Ges.m.b.H.**
Auhofstr 58, 1130 Vienna, Austria

Februari 1998

(SKI/RA report 023/97)

SKI Project Number 97194

This report concerns a study which has been conducted for the Swedish Nuclear Power Inspectorate (SKI). The conclusions and viewpoints presented in the report are those of the authors and do not necessarily coincide with those of the SKI.

Table of contents

1. INTRODUCTION

1.1 Background of the Project

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 NPPs, 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 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 CCIs, thus potentially neglecting an important risk contributor.

Considering the importance of CCI events and the fact that a systematic investigation of such events was never undertaken, **the SKI-RA's Mr. Ralph Nyman** initiated an activity aim of which is to identify actual occurrences of CCIs on the basis of international operational experience collected in the IRS data base. In particular, the aim of the project is to give a guidance on where more investigations may be warranted to enhance the considerations and the modelling of CCIs in PSA.

This report represents the final report of this phase of work, which was limited in both time and the scope to a minimum necessary to identify if the CCI deserves further investigation. This report summaries the results achieved during the project development, but also reflects the issues and the comments raised by the participants at the SKI's CCI seminar which was held in Stockholm on December 17, 1997. In particular, the seminar highlighted the importance of CCI and various partial failures which would lead to malfunctions of systems at plant. The seminar also concluded that the PSA consideration of CCI is a difficult issue, and that more guidance is necessary. The Seminar recognised that evaluation of operational events may be a way to define a minimum requirements for CCI treatment in PSAs.

1.2 Common cause initiators

Common Cause Initiators will give raise to increased frequency of initiating events as well as to unavailability of safety systems or of safety relevant operator response. They are relevant not only from probabilistic point of view but also can play important role in deterministic considerations as they may have impact on multiple safety barriers or defense in depth layers.
It should be noted that CCIs are often overlooked in the event analysis. A systematic in- depth analysis of operational events often concentrates on individual occurrences, not on full chain of occurrences. Root Cause Analysis focuses on areas/segments where remedial actions could be implemented.
The CCIs are not readily addressed in PSAs. The main attention is given to specific initiators provided in „generic“ lists such as EPRI list. Handling IE and system failures simultaneously is more complicated from methodological point of view.
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.

1.3 Objective

The objective of this project is to obtain practical insights relevant for the identification of Common Cause Initiators (CCIs) based on event data available in the NEA Incident Reporting System (IRS). The project is intended to improve the understanding of CCIs and, in consequence, their consideration in safety assessment of nuclear power plants and in particular plant specific probabilistic safety assessment.
It is also expected to provide some practical examples demonstrating safety importance of CCIs and help in determining the scope of farther investigation of this issue.

1.4 Scope

The project is a pilot study on CCI related issues. As such it is not expected to provide answers for all related questions. Its scope is limited to some practical insights that would help to improve the understanding of the issue and to establish directions for further work.
The project focus on identification of CCIs based on the existing operational experience accumulated in IRS. The following related issues are within the scope of this project:

- Determination of what type of information is essential in searching for CCIs;
- Gathering practical insights regarding CCI search strategy;
- Establishing a preliminary list of CCI event candidates in IRS database.

Other issues addressed in the project include:

- Comparison of CCI candidates with EPRI list of Initiating Events,
- Categorization of CCI candidates;
- Identification of CCI groups of highest concern;
- Determination if CCIs deserve further investigation and what should be the scope of investigation.

2. THE IRS REVIEW

2.1 Basic definitions

The following definition is used for Common Cause Initiators:
Common Cause Initiators are events that cause simultaneous (or consequential) occurrence of an Initiating Event (IE) and functionally degrade or disable systems(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.
An **Initiating Event** is a postulated event that creates a disturbance in a plant requiring some form of controlling or mitigating 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 to be pointed out that **mitigation systems** credited in PSAs are not limited to safety systems; in many cases they include safety related systems or normal operation systems. Therefore, the consideration of CCI events in the context of PSA should be broad enough in order to include dependent failures in all systems credited in PSAs (not only safety systems). These systems include also support systems required for successful operation of front line systems.
The concept of IE 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 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. Unrevealed 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.

2.2 Attributes of the database search

CCI definition discussed in Section 2.1 determine the basic attributes of the events that should be looked for in the event database. The following event attributes have to be investigated in order to identify CCIs:
(1) Effect of the event on plant operation;
(2) Degradation of safety significant systems;
(3) Failure type/mode of safety significant items.
Effect of the event. CCI candidates should involve an initiating event. Reactor scram 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 scram (automatic or manual).
Degradation of safety significant systems. 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.

2.3 Events of interest

Events that have the first two attributes (involve an IE and one or more failures that functionally degrade the mitigation systems associated with this IE) are of interest from the point of view of PSA. Two types of events can be distinguished within this group depending on the third attribute i.e. the type of accompanied failures (dependent or independent).

Events of the highest interest are events involving **consequential occurrences** (dependent on the specific IE). These events are real **Common Cause Initiators** that may have a considerable impact on the risk associated with this IE, particularly when they involve multiple failures. Neglecting these dependencies in a PSA model may lead to a considerable underestimation of the risk.

Events that involve **random occurrences** (independent of the specific IE) can be considered as **Accident Sequence Precursors** (AS Precursors). They should have been modeled in a PSA provided that it is detailed enough to include all the elements involved in the reported event. These events provide an interesting and useful possibility to verify the related risk estimates based on operational statistics. They also contribute to the IE frequency. However, these events are not relevant from the point of view of PSA model logic and do not provide information on dependencies.

It should be noted that some CCI events reported in IRS involve both types of occurrences i.e. consequential (dependent) and random (independent). In these events dependent degradation of systems is only a part of the whole degradation of the plant associated with the event.

In addition to CCI events and AS precursors there are other events that may provide useful insights related to CCI issue.

Events leading to degradation of systems that are not relevant from the point of view of accident mitigation may involve strong dependencies that in other conditions (or in plants with different design features) may lead to degradation of mitigation systems. These events may be considered as **potential CCI events**.

Another group of interest includes events that do not involve additional degradation of system but may provide useful insights regarding **potential dependency mechanisms**. For instance they may include common cause failures within the same system.

These two groups are also of interest for this study and will be covered in the review.

3. DATABASE SEARCH METHOD

3.1 Basic concept

Search attributes described in Section 2.2 are relevant in establishing a basic concept of event database search. Coding available in the IRS database make it possible to perform an automatic (computerized) for the presence of the reactor scram (attribute 1) and partially for the presence of degradation of safety significant items/systems (attribute 2).
Since the search for the attribute (1) cannot provide information on the type of the IE involved, a code based search for a degradation of safety significant items/systems (attribute 2) has limited capability 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.
However, a key word search may be useful as an additional way of checking the results of code based search for the degradation of certain systems. It is realized that the quality and completeness of data reported to the IRS may be sometimes questionable, particularly for events reported at the beginning of IRS operation.
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).

3.2 Code search

Two code-based searches were carried out within this step. In the first all events with the reactor scram were selected. In the second the list of events selected in the first search were reduced to events that involve at least one failure/degradation of safety significant systems.

- The codes used in the first search included:
  - Automatic reactor scram (code 6.1.1),
  - Manual reactor scram (code 6.1.2).
- The codes used in the second search included:
  - Degradation of items important to safety (code 1.2),
  - Failed/affected essential reactor auxiliary systems (code 3.B),
  - Failed/affected essential service systems (code 3.C),
  - Failed/affected electrical systems (code 3.E),
  - Failed/affected HVAC systems (code 3.H),
  - Failed/affected service auxiliary systems (code 3.K).

3.3 Key word search

A key-word search in the abstracts of the reports was applied in this step. 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.

3.4 Manual search

Manual search based on event description was applied to events identified through automatic search to eliminate events that were not CCI event candidates. The first step of this manual review was based on event titles to eliminate obvious cases.

Later elimination required a more careful analysis of each event based on event description. 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. Practically, all events had to be scrutinized very carefully in order to identify elements relevant from CCI point of view. In particular, the identification of dependencies between the IE and other occurrences was a difficult and time-consuming task.

Events that did not lead to degradation of additional plant systems or did not provide any insights regarding potential dependency mechanisms were eliminated from further analysis. Events that in addition to initiating event involved degradation of some systems were analyzed more carefully in order to identify and to classify dependencies involved.

4. RESULTS OF THE REVIEW

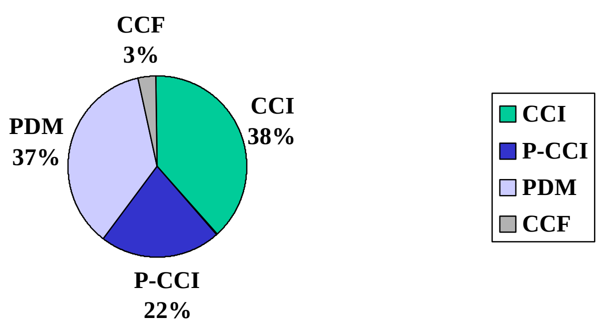
4.1 Overview of the search results

The initial automatic search (code based and key word based) reduced the number of events from over 2500 events included in IRS to about 400. These events were reviewed manually based on event titles in order to reduce further the list. Resulting selection included 153 events. These events are listed in Appendix A. Description of these events (report abstracts) are given in Appendix F.
This selection (153 events) was further analyzed manually based on the description of events provided in the report abstracts. Results of this analysis are summarized in Appendix B. Information provided for each event include description of initiating event and its direct cause, additional systems that were degraded and comments related to dependencies involved (or potential dependency mechanism).
There were 93 events that were eliminated from further analysis. These events do not involve any dependencies. They are simple initiating events or accident sequence precursors. The remaining 60 events 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 (CCIs),
- Common cause failures (CCFs),
- Events that indicate potential dependency mechanisms (PDMs).

Appendix C summarizes CCI related insights for these 60 events. This summary includes the event type, information on direct cause of the event and the type of dependencies involved.
Distribution of events among these event categories is given in Fig. 1.

FIG. 1. Ivents of interest identified in IRS database in relation to CCIs.



4.2 Event categorization

An attempt was made in order to categorize CCI event candidates. This categorization was performed for the final selection of events (60 events). The categorization takes into account the direct cause of the event and the dependency mechanism.
Description of the direct cause of the event includes malfunction type (human action or hardware failure) and the type of system/component involved in the direct cause.
The following types of malfunctions were used in this classification:

- Human interactions
- test related error
- operational error

- maintenance related error
- Hardware (component/system) failures
- Mechanical
- Electrical
- Instrumentation and Control.

FIG. 2. Systems and types of malfunctions involved in direct cause of events.

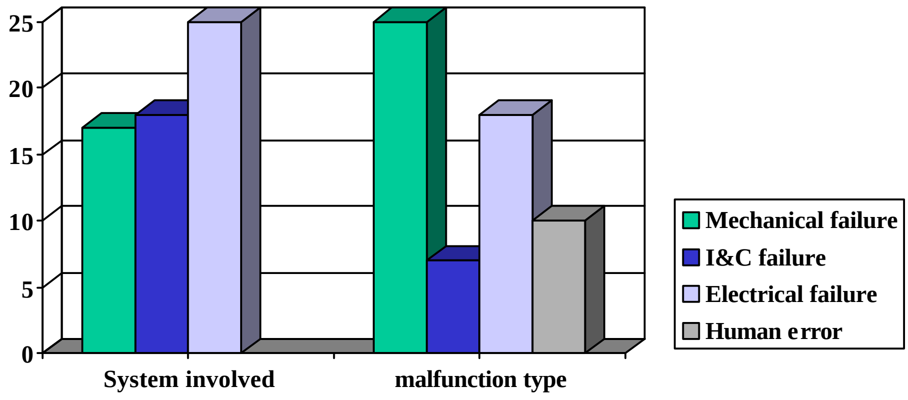


Fig. 2 shows distribution of direct causes among the above mentioned categories. As shown the distribution of events among system type categories (mechanical, electrical, I&C) is almost uniform. With regard to malfunction type involved about 80% of direct causes are hardware failures (sometimes in combination with human errors) and 20% human errors alone.

Dependency mechanisms identified include direct and indirect interrelations. Among direct dependency mechanisms the following dependency sources can be mentioned:

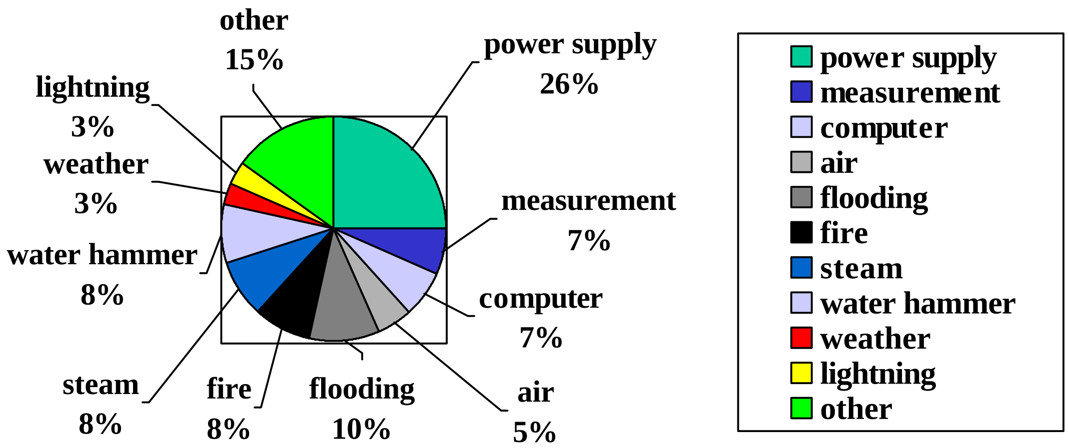
- Power supply
- Measurement/signal
- Computer support
- Instrument air.

Indirect dependency mechanisms include environmental factors, transients and external events:

- Flooding/spray
- Fire
- Steam environment
- Water hammer
- Lightning
- Cold weather.

Distribution of various dependency mechanisms among the events is shown in Fig. 3.

FIG. 3. Various dependency mechanisms involved in CCIs related events (60 events).



Among direct causes the dominating dependency mechanisms include electrical power supply and I&C (measurement/signal, computer support, instrument air) support functions. Among indirect mechanisms dominating mechanism include environmental area related dependencies (fire, flood, steam) and transients (water hammer).

4.3 Insights on CCI events identification

Direct cause of an event has been found of less importance with regard to event progression, extent of plant degradation and safety significance of the event. Malfunctions that induce common cause initiators may originate in electrical, mechanical and I&C systems. Direct causes involve mainly hardware failures but some of the events were induced by human errors.
Electrical power supply system and I&C systems (protection, control, indications) play important role in the events that involve dependent degradation of multiple safety systems. Related system malfunctions may be associated with failures within the systems but very often they are consequential response to failures originating in other systems (e.g. flood, fire, steam environment).
Several dependency mechanisms have been identified. The most typical chains of consequential occurrences are described below. This discussion may be helpful in focusing the analyst attention on plant areas that have highest potential to induce CCI events.
The most frequent dependent scenarios involve the electrical power supply system. In these scenarios failure and consequential degradation of the electrical power supply leads to turbine trip followed by the reactor trip (IE) and at the same time results in malfunctions in the protection and control systems that provide essential safety functions to several accident mitigation (front line) systems.
Typical examples of such degradations are loss of AC power automatic transfer, emergency diesel automatic start, automatic load shedding (e.g. event 0437G5) or spurious opening of safety relief valves (e.g. event 059400). This dependency mechanism was observed in several events (120602, 103500, 0437G5, 042503).
Failure in the electrical power supply system may originate within the system or may be a consequence of failures in other systems e.g. spraying the electrical cabinets due to a spurious actuation of fire extinguishing system (event 059400).
Similar dependency mechanism involve failure of electrical power supply that leads to turbine trip followed by the reactor trip (IE) and degrades an essential support system(s) common to several accident mitigation systems such as service water system (e.g. event 146400).

4.4 Comparison with EPRI list of IE

For each of the events selected as CCI related initiating events were defined and compared with EPRI IE list. The results of this task are documented in Appendix D.
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.
In certain cases assignment of IE based on the ‘heat production-removal balance’ definition of IE was difficult. One of the exceptions from this approach was the use of an EPRI category „Loss of power supply to necessary equipment“ (37).

4.5 Analysis of selected examples

Example analysis of selected CCI related events is provided in Appendix E. The results of this analysis are summarized in the form of tables that provide the most important information needed for classifying the event.

5. CONCLUSIONS

The following conclusions were reached:

- Review of IRS events confirmed that CCI events are observed in operational history of nuclear power plants. In addition to real CCIs that are not very frequent, there are potential CCIs that depending on plant conditions could lead to real CCIs.
- CCI events observed in operational history are usually very complex events involving many occurrences consequential and random. Only some of these occurrences lead to dependent degradation of mitigation systems.
- Dependent degradation of mitigation systems associated with CCI events identified in the project is usually limited. Partial failures (degradation) of system functions are typical. Loss of automatic features or safety related indications are examples of such degradations.
- Identification of CCI candidates is difficult and time consuming. Application of automatic search in the database is limited. Manual search based on detailed analysis of event description is important element of the review.
- In most cases event descriptions provided in IRS report abstracts were sufficient to classify the events as CCI candidate. However in some cases report abstract did not contain information of sufficient detail. The most difficult part of the analysis was the identification of consequential failures.
- Direct interrelations between the accident mitigation systems through common support systems are dominant dependency mechanisms involved in the CCI events. The most important contributors of this type are electrical power supply systems and I&C systems. Environmental and area related events such as fire, flooding, water spray and steam have also been found to be important sources of dependency.
- Majority of the above mentioned dependencies are plant specific and determined by plant design features. The analyst can identify them and model explicitly in a PSA. Modeling of these dependencies requires a detailed analysis of all system interrelations. It should be noted that the methodology for area related events such as fire and flooding is relatively well established.
- In addition to the above mentioned dependencies the IRS review identified other potential dependency mechanisms that have not been modeled in PSAs and that are more difficult to be consider explicitly in the plant logic model. Transients (e.g. water hammer, grid disturbances) and external events (e.g. lightning, cold weather) are examples of such dependency mechanisms. Another dependency mechanism, difficult to be modeled, is related to human factors. Such issues as non-conservative planning of maintenance or errors of commission have been found in the review as one of the possible sources of dependency.
- The review performed within the framework of this project 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.
- The overall conclusion is that CCI event are an important risk contributors. At the same time, those events (due to modelling difficulties) are less extensively analysed that it would be warranted.

- **Further investigation is strongly recommended, initially to focus on the following CCI groups:**
    - **Electrical system failures and, in particular, partial failures**
    - **Localised floods and spraying of electric elements**
    - **DC supply system**
    - **Instrumentation and control, in particular partial failures**

- **The analysis of operational events is capable of providing the guidance on how to model CCI events in PSAs. Systematic analysis could lead to a systematic definition of basic modelling requirements which could serve as a basis for a more formal PSA modelling guidance (a chapter of a PSA guide).**

APPENDIX A

EVENTS SELECTED FOR DETAILED ANALYSIS – EVENT TITLES

REPORT	PLANT CODE	INCIDENT DATE	EVENT TITLE
000200	CH-4	80.02.06	INADVERTENT CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
000300	US-302	80.02.26	LOSS OF REACTOR COOLANT FROM THE HIGH PRESSURE REACTOR COOLANT SYSTEM
001104	US-296	80.06.28	PARTIAL FAILURE OF THE SCRAM SYSTEM
0014G5	FI-3	79.08.29	PIPE RUPTURE IN THE REACTOR WATER CLEAN-UP SYSTEM
0014G8	FI-4	80.02.22	RAPID TEMPERATURE DECREASE OF REACTOR PRESSURE VESSEL DUE TO OPERATING ERROR
0019G5	FR-20	80.04.09	INHIBITION OF SAFETY INJECTION AFTER SPONTANEOUS OPENING OF THE PRESSURISER SPRAY REGULATION VALVE
004002	US-336	81.01.02	LOSS OF 125V DC BUS.
004100	US-296	81.05.22	BROWNS FERRY 3 REACTOR SITE ALERT DUE TO HIGH LEAKAGE INTO DRYWELL
006200	ES-6	81.01.25	SPURIOUS SAFETY INJECTION DUE TO HIGH DIFFERENTIAL PRESSURE BETWEEN STEAM LOOPS.
006600	SE-5	80.02.03	INADVERTENT SAFETY INJECTION.
007200	US-321	81.06.26	HIGH PRESSURE COOLANT INJECTION SYSTEM'S FAILURE TO AUTOMATICALLY START FOLLOWING REACTOR TRIP
007800	ES-6	81.05.08	ACTUATION OF SAFETY INJECTION DUE TO ADJUSTMENT FAILURE OF STEAM FLOW TRANSMITTERS.
009702	US-206	81.09.03	FAILURE OF HIGH PRESSURE SAFETY INJECTION SYSTEM
011800	JP-20	81.11.22	MANUAL STOP DUE TO TROUBLE IN THE MAIN FEEDWATER CONTROL VALVE IN THE REACTOR FEEDWATER SYSTEM
012500	DE-9	81.03.04	PARTIAL FAILURE OF THE THREE PHASE SUPPLY SYSTEM.
012600	JP-9	82.02.14	MALFUNCTION OF MASTER CONTROLLER IN FEED WATER CONTROL SYSTEM.
014800	US-339	81.07.03	FIRE RESULTING FROM TRANSFORMER FAILURE.
016500	US-260	81.03.13	REACTOR SCRAM AND LOSS OF REDUNDANT SAFETY SIGNALS.
0176G4	US-244	81.11.14	INADVERTENT ACTUATION OF FIRE SUPPRESSION SYSTEM.
019000	JP-5	82.07.24	REACTOR SCRAM DUE TO MAIN STEAM ISOLATING VALVE CLOSURE.
021600	CA-4	82.10.10	LOSS OF REGULATION INCIDENT.
0218G2	US-270	82.06.28	STEAM EROSION IN TURBINE EXHAUST LINES
0218G3	US-271	82.01.27	LEAKAGE FROM MOISTURE SEPARATOR DRAIN LINE
0218G4	US-344	85.03.09	EROSION AND RUPTURE OF HEATER DRAIN PIPING, PARTLY DUE TO MISUSE OF PUMP DISCHARGE PIPE
0218G5	US-213	85.03.16	FEEDWATER LINE RUPTURE DUE TO EROSION IN AREA NOT REGULARLY INSPECTED
022100	BE-2	82.08.04	BLACKOUT AFFECTING THE DOEL POWER STATION
022900	NL-2	83.01.03	LEAKAGE IN THE CHEMICAL AND VOLUME CONTROL SYSTEM
023100	BE-6	82.11.05	FAILURE OF ALL THREE SEALS ON A PRIMARY PUMP
023600	US-321	82.07.03	FAILURE OF ELEVEN SAFETY RELIEF VALVES TO ACTUATE AT SETPOINT.
0241G1	US-369	82.01.11	FROZEN INSTRUMENTATION LINES CAUSE INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES.
024702	BE-3	83.04.13	PRIMARY PUMP SEAL FAILURE
025800	US-265	82.06.22	LOSS OF AUXILIARY POWER AFFECTS TWO UNITS.
026200	JP-17	82.12.20	MALFUNCTION OF MAXIMUM FLOW LIMITER IN FEED WATER CONTROL CIRCUIT
026300	JP-13	82.12.24	REACTOR TRIP DUE TO INADVERTENT OPERATION ON POWER SUPPLY TO THE CONTROL ROD DRIVE MECHANISM
027002	IT-4	83.03.29	DRAIN OF PRIMARY WATER DUE TO MISALIGNMENT OF A RHR VALVE
029000	US-324	82.10.10	EMERGENCY BUS LOSS DUE TO BREAKER PROBLEMS
029600	US-219	82.12.15	REACTOR SCRAM DUE TO EXCESSIVE VALVE LEAKAGE
030302	JP-21	83.02.18	TROUBLE WITH ELECTRICAL SUPPLY SYSTEM CAUSED BY LIGHTNING
031900	DE-15	82.06.06	IMPACT OF A LIGHTNING STROKE INTO PLANT'S 220 KV LINE
032100	DE-15	83.04.15	SHORT CIRCUIT IN THE STATION SERVICE SUPPLY CAUSED BY HUMAN ERROR AND SUBSEQUENT FAILURE OF A STATION SERVICE
032700	US-309	83.01.25	MAIN FEEDWATER LINE BREAK DUE TO WATER HAMMER.
033000	US-318	83.02.03	INADVERTENT RPS TRIP WITH PORV ACTUATION.
035700	JP-12	83.09.02	LEAKAGE FROM THE SEATS OF PRESSURIZER RELIEF VALVES
035800	JP-2	83.09.04	REACTOR SCRAM DUE TO LOSS OF DC POWER SUPPLY.
037500	US-366	82.08.25	UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT
039000	US-254	83.09.15	REACTOR DEPRESSURIZATION RESULTING FROM FAULTY STEAM RELIEF VALVE.
039600	DE-9	83.10.21	NON-CLOSURE OF SAFETY RELIEF VALVE.
040600	JP-5	83.11.19	REACTOR SCRAM DUE TO MAIN STEAM ISOLATION VALVES CLOSURE
042503	FR-16	84.04.14	VOLTAGE DROP FOLLOWED BY LOSS OF TRAIN A 48 VOLT BUS LEADING TO LOSS OF BOTH OFF-SITE POWER AND TRAIN A DIESEL G
0437G3	US-388	84.07.26	TEMPORARY LOSS OF OFFSITE AND ONSITE AC ELECTRICAL POWER.
0437G5	US-388	84.07.26	TESTING RESULTS IN STATION BLACKOUT
044300	US-311	84.04.06	WATER HAMMER IN FEEDWATER PIPING AND SUBSEQUENT SCRAM DUE TO FEEDWATER SYSTEM PROBLEMS
044400	US-254	84.03.05	REACTOR SHUTDOWN DUE TO INOPERABLE HPCI SYSTEM AND SAFETY RELIEF VALVE FAILURE.
049500	US-271	84.04.20	HIGH PRESSURE COOLANT INJECTION SYSTEM LOCKOUT.
051604	US-346	85.06.09	LOSS OF MAIN AND AUXILIARY FEEDWATER SYSTEMS
052000	CA-11	85.01.03	LOSS OF ELECTRICAL POWER TO A UNIT BA-85-2
052300	US-331	84.11.04	EXPLOSION AND FIRE IN AUXILIARY TRANSFORMER RESULTS IN LOSS OF STARTUP TRANSFORMER
052600	US-483	84.11.	REACTOR TRIP DUE TO AIR LINE FAILURES CAUSED BY FATIGUE CRACKING
053200	US-269	84.12.02	ANTICIPATORY REACTOR TRIP ON GENERATOR FIELD BREAKER TRIP CAUSED BY WIRE IN AMPHENOL CONNECTOR
053900	US-369	84.08.21	SWITCHYARD COMPUTER DESIGN DEFICIENCY CAUSES LOSS OF NORMAL OFF-SITE POWER.
055000	SE-14	85.05.03	REACTOR TRANSIENT CAUSED BY GENERATOR BREAKER FAILURE
0559G2	US-416	85.02.10	REACTOR SCRAM ON LOW CONDENSER VACUUM WITH SUBSEQUENT MSIV FAILURES
056300	US-321	85.01.16	INOPERABLE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) FOLLOWING LOW VES
056600	US-265	85.01.16	REACTOR SCRAM ON LOSS OF CONDENSER VACUUM DUE TO FAILED EXPANSION JOINT, AND SUBSEQUENT HIGH CONTAINMENT P
057000	US-247	84.12.28	INOPERABLE SAFETY INJECTION PUMPS
0572G0	US-219	85.06.12	UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT
058803	US-206	85.11.21	LOSS OF ALL IN-PLANT AC POWER, REACTOR TRIP AND WATER HAMMER AT SAN ONOFRE-1
059400	US-321	85.05.15	SYSTEM INTERACTION EVENT RESULTING IN REACTOR SYSTEM SAFETY RELIEF VALVE OPENING FOLLOWING A FIRE PROTECTION
060100	CA-11	85.06.23	SHUTDOWN CAUSED BY FAILURE OF NEUTRON OVERPOWER DETECTORS
061000	US-373	85.05.31	FLOODING RESULTS FROM EXPANSION JOINT FAILURE AND INSTALLATION ERROR
061600	US-334	85.08.29	SAFETY INJECTION AND REACTOR TRIP DUE TO LOSS OF STATION INSTRUMENT AIR PRESSURE.
061800	US-413	85.08.15	BLACKOUT SIGNAL AND INTERACTION EVENT BETWEEN UNITS
0630G1	US-528	85.10.03	LOSS OF OFFSITE POWER CAUSED BY PROBLEMS IN FIBER OPTICS SYSTEMS
063100	US-316	85.10.29	REACTOR TRIP, POSSIBLY DUE TO SHORT PHOTOHELIC CELL IN OUTPUT POWER SUPPLY, WITH ONE REACTOR TRIP BREAKER FAILI
063300	US-302	85.10.26	REACTOR TRIP RESULTS FROM ERRONEOUS CONTROL BOARD INFORMATION DUE TO INVERTER FAILURE.
064100	IT-4	86.02.15	INADVERTENT OPENING OF A RELIEF VALVE DUE TO A SPURIOUS CONTACT BETWEEN WIRES INSIDE AN ELECTRIC PENETRATION
064700	US-528	85.12.16	INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION AND SUBSEQUENT REACTOR TRIP
064800	US-220	85.11.01	LOSS OF INSTRUMENT AIR CAUSES REACTOR SCRAM WITH HIGH PRESSURE COOLANT INJECTION SUBSEQUENTLY FAILING.
064900	US-293	86.04.04	RECURRENT SPURIOUS CONTAINMENT ISOLATION, MAIN STEAM ISOLATION VALVE FAILURES TO REOPEN AND PRESSURIZATION O
067600	DE-22	86.04.15	COMPLETE LOSS OF AUXILIARY FEEDWATER CONTROL FOR BOTH STEAM GENERATORS
067800	ES-10	85.02.05	WATER DRIPPING IN CONTROL ROOM.
068000	ES-1	85.09.	SPURIOUS OPENING OF PRESSURIZER SPRAY VALVE
069300	US-458	86.01.07	UNLABELED SWITCH RESULTS IN INADVERTENT ACTUATION OF DELUGE SPRAY SYSTEM AND SUBSEQUENT SCRAM
074700	US-261	86.01.28	LOSS OF OFF-SITE POWER DUE TO UNNEEDED ACTUATION OF STARTUP TRANSFORMER PROTECTIVE DIFFERENTIAL DELAY
0785G0	US-GEN	82. .	OPERATIONAL EXPERIENCE INVOLVING LOSSES OF ELECTRICAL INVERTERS
0819G0	US-GEN	87. .	DEPRESSURIZATION OF REACTOR COOLANT SYSTEM IN PRESSURIZED WATER REACTORS
086100	DE-27	87.09.28	FALSE TRIGGERING OF REACTOR PROTECTION SIGNALS DUE TO A FAILED CLOCK GENERATOR
086300	DE-18	88.04.19	DAMAGE OF AN INSTRUMENT TRANSFORMER CAUSES LOSS OF OFF-SITE POWER TO BOTH UNITS
087400	US-324	88.01.02	FAILURE OF TWO SETS OF REDUNDANT PRIMARY CONTAINMENT ISOLATION VALVES AT BRUNSWICK UNIT 2
087700	US-374	88.03.09	LOSS OF RECIRCULATION PUMPS ACCOMPANIED BY SEVERE POWER OSCILLATIONS AT LASALLE UNIT 2

REPORT	PLANT CODE	INCIDENT DATE	EVENT TITLE
088200	NL-2	87.10.10	SHORT CIRCUIT IN MAIN TRANSFORMER CAUSING REACTOR TRIP, ELECTRICAL FAILURES AND SECONDARY SIDE TRANSIENTS
088400	BE-8	87.10.13	REACTOR TRIP, FOLLOWED BY AN ECCS ACTUATION, DURING A QUALIFICATION TEST OF A MODIFICATION ON A SECOND LEVEL PI
089700	CA-7	88.02.06	REACTOR TRIP DUE TO FAILURE OF REGULATING SYSTEM IN-CORE FLUX DETECTORS
091400	US-414	88.03.09	FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS
093900	GB-20B	87.10.10	TRIP OF REACTOR 2 DUE TO LOSS OF 11 KV UNIT BOARD
095800	FR-7	87.01.12	PARTIAL BLOCKAGE OF THE WATER INTAKE OF ONE UNIT, AND LOSS OF OFF-SITE POWER TO THE TWIN DURING COLD WEATHER
102502	ES-3	89.10.19	FIRE IN ONE TURBINE GENERATOR GROUP AND SUBSEQUENT FAILURE OF SAFETY SYSTEMS
103500	US-316	89.08.14	REACTOR TRIP DUE TO UNDERVOLTAGE ON CONTROL ROOM INSTRUMENTATION DISTRIBUTION PANEL AT D.C. COOK UNIT 2
1046G0	US-GEN	. .	ELECTRICAL BUS BAR FAILURES
108802	US-424	90.03.20	LOSS OF VITAL AC POWER WITH SUBSEQUENT REACTOR COOLANT SYSTEM HEAT-UP AT VOGTLE UNIT 1
108803	US-424	90.03.20	LOSS OF VITAL AC POWER AND THE RESIDUAL HEAT REMOVAL SYSTEM DURING MID-LOOP OPERATION AT VOGTLE UNIT 1 (FINAL
109500	US-237	90.01.16	LOSS OF OFF-SITE POWER WITH MULTIPLE EQUIPMENT FAILURES AT DRESDEN UNIT 2
1109G0	US-GEN	. .	POTENTIAL FAILURES OF ROSEMOUNT PRESSURE AND DIFFERENTIAL PRESSURE TRANSMITTERS DUE TO LOSS OF FILL-OIL
115200	JP-25	90.01.02	REACTOR MANUAL SHUTDOWN DUE TO MISINDICATION OF PRIMARY LOOP RECIRCULATION PUMP MOTOR LOWER BEARING OIL L
1180G0	US-GEN	. .	SOLENOID-OPERATED VALVE PROBLEMS AT U.S. POWER REACTORS - OPERATING EXPERIENCE FEEDBACK REPORT
120600	US-410	91.08.13	REACTOR SCRAM FOLLOWING LOSS OF FIVE UNINTERRUPTIBLE POWER SUPPLIES AND PARTIAL LOSS OF CONTROL ROOM INSTRU
120602	US-410	91.08.13	REACTOR SCRAM FOLLOWING LOSS OF FIVE UNINTERRUPTIBLE POWER SUPPLIES AND PARTIAL LOSS OF CONTROL ROOM INSTRU
1326G0	US-220	00. .	RECENT LOSS OR DEGRADATION OF SERVICE WATER SYSTEM
136300	ES-2	92.09.14	REACTOR SCRAM DUE TO LOW PRESSURE SIGNAL OF THE INSTRUMENT AIR
1446G0	US-410	91.08.13	INADEQUATE MAINTENANCE OF UNINTERRUPTIBLE POWER SUPPLIES AND INVERTERS
146204	CA-5	94.12.10	REACTOR COOLANT LEAKAGE AND EMERGENCY COOLANT INJECTION DUE TO A RELIEF VALVE FAILURE AND PIPE BREAK
146400	CA-11	94.03.02	INTERRUPTION OF 48V DC CLASS I POWER RESULTS IN A LOSS OF UNIT LOW PRESSURE SERVICE WATER (LPSW) AND A PARTIAL LO
1493G0	US-413	93.02.25	PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1993 - A STATUS REPORT (NUREG/CR-4674, ORNL/NOAC-232, Vol. 1
1553G0	US-339	93.04.16	MALFUNCTION IN MAIN GENERATOR VOLTAGE REGULATOR CAUSING OVERVOLTAGE AT SAFETY- RELATED ELECTRICAL EQUIPM
601200	IN-1	83.12.17	INSTRUMENT AIR FAILURE AT TAPS
602300	KR-1	84.11.13	REACTOR TRIP ON ROD CONTROL SYSTEM FAILURE
603900	HU-1	85.03.29	EMERGENCY FEEDWATER SUPPLY ACTUATED
604700	SU-30	84.05.11	SHUTDOWN OF THE REACTOR DUE TO OPENING AND NON-CLOSURE OF A PILOT-OPERATED RELIEF VALVE
605400	CS-13	85.01.03	RCS OVERCOOLING RESULTING FROM A FAILURE TO CLOSE TURBINE BYPASS VALVES
607600	IN-5	86.06.25	STUCK OPEN INSTRUMENTED RELIEF VALVE IN MAPS UNIT NO. 1
608600	SU-8	87.08.03	POWERING DOWN OF NO. 2 REACTOR AT NOVOVORONEZH NPP ON 08.03.87 OWING TO A FAULT IN THE CONTROL SYSTEM DRIVE M
609500	SU-18	82.10.15	DISRUPTION OF NORMAL OPERATING CONDITIONS AND EMERGENCY SHUT-DOWN OF THE FIRST UNIT OF THE VVER REACTOR OF T
609600	YU-1	84.01.09	PRESSURIZER SPRAY CONTROL VALVE FAILURE AND DEPRESSURIZATION OF REACTOR COOLANT SYSTEM
611500	SU-56	87.12.11	LOSS OF OFF-SITE POWER WITH NON-AVAILABILITY OF EMERGENCY POWER SUPPLY
611700	SU-97	88.02.08	DAMAGE TO FAST ACTING GATE VALVE OF THE PRESSURIZER WATER INJECTION SYSTEM
612100	AR-1	85.12.25	LOSS OF POWER IN 6.6 KV EMERGENCY BARS
612400	CS-5	86.02.01	INADVERTENT ACTUATION OF SAFETY SYSTEMS RESULTING IN SCRAM
613600	SU-96	88.03.27	OPENING OF ALL FAST-ACTING STEAM DUMP VALVES AT A SPURIOUS SIGNAL AND FAILURE OF THE TURBINE CONTROL SYSTEM
614300	SU-34	87.07.30	NON-CLOSURE OF MAIN SAFETY VALVE DURING REGULATORY TESTING AND ADJUSTMENT PRIOR TO START-UP
614700	SU-78	88.01.30	FAILURE OF THE PRESSURIZER INJECTION VALVE
614800	SU-97	88.03.22	REACTOR SCRAM DUE TO LEAK IN THE PRESSURE INSTRUMENTATION LINE OF THE PRESSURIZER LEVEL TRANSMITTER
616300	SU-28	88.12.08	SPURIOUS OPENING OF PRESSURIZER RELIEF VALVE
616400	SU-47	88.09.05	UNPLANNED SHUTDOWN OF IGNALINSKAYA 2 ON SPURIOUS SIGNALS DUE TO CABLE FIRES
617400	SU-96	89.01.04	ACTUATION OF THE FIRE FIGHTING SYSTEM AND ONE CHANNEL OF THE SAFETY SYSTEM DUE TO SPURIOUS SIGNALS
620400	SU-44	89.02.06	LOSS OF POWER IN THE ACTUATION CIRCUITS FOR POWER SUPPLY TO THE REACTOR CONTROL AND PROTECTION SYSTEM
6255G0	CS-GEN	90. .	GENERIC PROBLEM OF LOAD REJECTIONS
6256G0	CS-GEN	90.12.04	DISCONNECTION OF ALL DUKOVANY UNITS FROM GRID.
6274G0	SU-GEN	90. .	UNPLANNED SHUTDOWNS OF VVER-1000 PLANTS DUE TO DESIGN DEFICIENCIES IN SERVICE WATER SYSTEMS FOR ESSENTIAL EQU
627700	SU-39	91.04.10	ACTIVATION OF THE EMERGENCY REACTOR PROTECTION SYSTEM DUE TO SPURIOUS SIGNAL DURING DEENERGIZATION OF THE SK
630400	KR-6	91.01.03	INADVERTENT REACTOR COOLANT SYSTEM DEPRESSURIZATION DUE TO PRESSURIZER SPRAY VALVE FAILURE
633100	SU-97	91.12.27	BATTERY MALFUNCTION DETECTED DURING TESTING
6339G2	SU-GEN	93.02.02	TOTAL LOSS OF POWER AT KOLA NPP UNITS CAUSED BY GRID DISTURBAN- CES DUE TO TORNADO
634400	SU-96	92.09.03	DEFICIENCIES IN THE ORGANIZATION OF STAND-BY DIESEL AND SAFETY SYSTEM BATTERY OPERATION
635000	RU-12	92.11.17	DE-ENERGIZATION OF DC SWITCHBOARD DUE TO DAMAGE TO A REVERSIBLE MOTOR GENERATOR AND BATTERY TRIPPING
635400	RU-17	93.03.21	REDUCTION IN FEEDWATER FLOW RATE OWING TO OPERATOR ERROR AND DEFICIENCIES IN PROCEDURE
635900	RU-32	93.05.27	UNAUTHORIZED ACTUATION OF A PRESSURIZER PILOT-OPERATED RELIEF VALVE
636000	RU-39	93.03.04	REACTOR SCRAM ON HI-HI STEAM DRUM LEVEL DUE TO DEENERGIZATION OF THE 0.4 KV BUS
636200	RU-22	93.02.17	REACTOR COOLANT PUMP TRIPPING AT REDUCED SEALING WATER SUPPLY DUE TO FILTER CLOGGING
640300	CZ-8	94.06.14	ACTUATION OF THE REACTOR POWER LIMITATION SYSTEM AND CONSEQUENT REACTOR MANUAL SCRAM
640400	CZ-5	94.09.22	MANUAL REACTOR TRIP AS A RESULT OF FLOODING OF REACTOR CONTROL AND PROTECTION SYSTEM ROOM
641200	RU-30	94.05.13	LOSS OF UNIT AUXILIARY POWER AND LOSS OF POWER TO ESSENTIAL LOADS (CATEGORY 1 RELIABILITY)
702500	US-237	90.08.02	ONSITE ANALYSIS OF THE HUMAN FACTORS OF AN EVENT AT DRESDEN UNIT 2 ON AUGUST 2, 1990 (SPURIOUS SAFETY RELIEF VALV
703200	SK-2	95.08.17	REACTOR SCRAM DUE TO STEAM GENERATOR LEVEL SIGNAL DROP FOLLOWING NaOH INGRESS INTO TURBINE CONDENSATE LINE

APPENDIX B

DETAILED ANALYSIS SUMMARY

REPORT	TITLE	IE TYPE	IE DIRECT CAUSE	ADD DEG
000200	INADVERTENT CLOSURE OF ALL MAIN STEAM ISOLATION VALVES	Transient Closure of MSIVs	Test related human error	Cond
001104	PARTIAL FAILURE OF THE SCRAM SYSTEM	Transient Manual scram	I&C deficiency	RPS
004002	LOSS OF 125V DC BUS.	Transient Loss of DC bus	Breaker failure	Sever Indica
006200	SPURIOUS SAFETY INJECTION DUE TO HIGH DIFFERENTIAL PRESSURE BETWEEN STEAM LOOPS.	Transient Inadvertent safety injection	Misaligned valve	SGs c prote
009702	FAILURE OF HIGH PRESSURE SAFETY INJECTION SYSTEM	Transient - Loss of AC power to necessary systems	RPS and Control & Indication System	MFW and c
012500	PARTIAL FAILURE OF THE THREE PHASE SUPPLY SYSTEM.	Transient - Loss of power to necessary systems	Maintenance related human error	RPS,
014800	FIRE RESULTING FROM TRANSFORMER FAILURE.	Transient - Loss of load due to transformer damage	Transformer fire due to electrical fault	Electr
016500	REACTOR SCRAM AND LOSS OF REDUNDANT SAFETY SIGNALS.	Transient – inadvertent reactor trip	False signal due to valve misalignment	Sever safety
0176G4	INADVERTENT ACTUATION OF FIRE SUPPRESSION SYSTEM.	Transient – manual reactor trip	Inadvertent actuation of fire sprays	RPS,
021600	LOSS OF REGULATION INCIDENT.	Transient Increase of reactor power	Failure of the reactor control computer	No
0218G2	STEAM EROSION IN TURBINE EXHAUST LINES	Transient – Manual reactor trip	Steam line rupture due to erosion	Non-s electr
0218G4	EROSION AND RUPTURE OF HEATER DRAIN PIPING, PARTLY DUE TO MISUSE OF PUMP DISCHARGE PIPE	Transient – Turbine trip	Spurious turbine protection signal	FW a secon
0218G5	FEEDWATER LINE RUPTURE DUE TO EROSION IN AREA NOT REGULARLY INSPECTED	Transient – Manual reactor trip	FW pipe rupture	Poten motor
022100	BLACKOUT AFFECTING THE DOEL POWER STATION	Transient – Loss of off-site power	Spurious protection signal during TG test	Emer (com
022900	LEAKAGE IN THE CHEMICAL AND VOLUME CONTROL SYSTEM	Transient – Manual reactor trip	Leakage from a valve in CVCS	Non-s equip
0241G1	FROZEN INSTRUMENTATION LINES CAUSE INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES.	Transient – ESFAS spurious actuation	Frozen instrumentation lines	No
025800	LOSS OF AUXILIARY POWER AFFECTS TWO UNITS.	Transient – Loss of off-site power	Maintenance related human error	On-si suppl
027002	DRAIN OF PRIMARY WATER DUE TO MISALIGNMENT OF A RHR VALVE	Transient – Interface LOCA	Misalignment of the valve, maintenance error	RHR
029600	REACTOR SCRAM DUE TO EXCESSIVE VALVE LEAKAGE	Transient – Manual reactor trip	Feedwater piping vibrations	MFW to wa
030302	TROUBLE WITH ELECTRICAL SUPPLY SYSTEM CAUSED BY LIGHTNING	Transient – Increase in feedwater flow	I&C faults due to lightning	RPS a (I&C
031900	IMPACT OF A LIGHTNING STROKE INTO PLANT'S 220 KV LINE	Transient – Generator trip	Lightning struck the main external electrical line	TG co safety
032700	MAIN FEEDWATER LINE BREAK DUE TO WATER HAMMER.	Transient – Loss of feedwater	Spurious reactor trip, motor driven FWPs out of service	SG fe dama
037500	UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT	Transient – Closure of MSIV	Valve disk separation	RCIC failur
042503	VOLTAGE DROP FOLLOWED BY LOSS OF TRAIN A 48 VOLT BUS LEADING TO LOSS OF BOTH OFF-SITE POWER AND TRAIN A DIESEL GENERATOR	Transient - Loss of DC power to necessary systems	Control card failure	Off-s start,
0437G5	TESTING RESULTS IN STATION BLACKOUT	Transient - Loss of DC power to necessary systems	Test related human error	EPS (load s
044300	WATER HAMMER IN FEEDWATER PIPING AND SUBSEQUENT SCRAM DUE TO FEEDWATER SYSTEM PROBLEMS	Transient – feedwater flow disturbances	FW regulation valves testing, check valve failure	MFW
049500	HIGH PRESSURE COOLANT INJECTION SYSTEM LOCKOUT.	Transient – MSIV closure	Pilot valve failure	RCIC not cl
052300	EXPLOSION AND FIRE IN AUXILIARY TRANSFORMER RESULTS IN LOSS OF STARTUP TRANSFORMER	Transient – Failure of plant auxiliary transformer	Short circuit followed by fire	EPS (suppl
053900	SWITCHYARD COMPUTER DESIGN DEFICIENCY CAUSES LOSS OF NORMAL OFF-SITE POWER.	Transient – transmission line power breaker trip	Switchyard control computer fault (HE)	No
057000	INOPERABLE SAFETY INJECTION PUMPS	SI pumps found to be inoperable	Boric acid solidification, binding due to gas	All SI
058803	LOSS OF ALL IN-PLANT AC POWER, REACTOR TRIP AND WATER HAMMER AT SAN ONOFRE-1	Transient – Manual reactor trip	Failure of auxiliary transformer	EPS (MFW
059400	SYSTEM INTERACTION EVENT RESULTING IN REACTOR SYSTEM SAFETY RELIEF VALVE OPENING FOLLOWING A FIRE PROTECTION DELUGE SYSTEM MALFUNCTION.	Transient – Manual reactor trip	Instrument water line damaged	HPCI spray
061000	FLOODING RESULTS FROM EXPANSION JOINT FAILURE AND INSTALLATION ERROR	Transient – Loss of circulating water	CW pump flexible expansion joint rupture	Non e water
061600	SAFETY INJECTION AND REACTOR TRIP DUE TO LOSS OF STATION INSTRUMENT AIR PRESSURE.	Transient – MSIVs closure	IA pressure low due to failed solder fitting	SI,(p RHR

REPORT	TITLE	IE TYPE	IE DIRECT CAUSE	ADD DEG
061800	BLACKOUT SIGNAL AND INTERACTION EVENT BETWEEN UNITS	Transient – Loss of off-site power	Test related human error	CVC cooli
063300	REACTOR TRIP RESULTS FROM ERRONEOUS CONTROL BOARD INFORMATION DUE TO INVERTER FAILURE.	Transient – Manual turbine trip	Inverter failure, loss of power supply	MFW
064100	INADVERTENT OPENING OF A RELIEF VALVE DUE TO A SPURIOUS CONTACT BETWEEN WIRES INSIDE AN ELECTRIC PENETRATION	Transient – Inadvertent openning of a relief valve	Spurious contact between wires in a penetration	No
064700	INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION AND SUBSEQUENT REACTOR TRIP	Transient – SG low level	Failure of ventilation fan in BOP ESFAS	EPS, AFW
064800	LOSS OF INSTRUMENT AIR CAUSES REACTOR SCRAM WITH HIGH PRESSURE COOLANT INJECTION SUBSEQUENTLY FAILING.	Transient – Increase of FW	IA service related error	FCV, valve
067800	WATER DRIPPING IN CONTROL ROOM.	Transient – Manual reactor scram	CR flood from fire deluge system in the cable room	Spuri
069300	UNLABELED SWITCH RESULTS IN INADVERTENT ACTUATION OF DELUGE SPRAY SYSTEM AND SUBSEQUENT SCRAM	Transient – Increase of reactor pressure	Flood (actuation of fire deluge system)	Turbi relate
086100	FALSE TRIGGERING OF REACTOR PROTECTION SIGNALS DUE TO A FAILED CLOCK GENERATOR	Transient – False RPS signals, low SG water level	Short circuit in a clock generator	MFW
091400	FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS	Transient – Instability in feedwater flow	FCV clogged with Asiatic clam shells	AFW
095800	PARTIAL BLOCKAGE OF THE WATER INTAKE OF ONE UNIT, AND LOSS OF OFF-SITE POWER TO THE TWIN DURING COLD WEATHER	Transient – Loss of circulating water	Ice floes blocked cooling water intake	
102502	FIRE IN ONE TURBINE GENERATOR GROUP AND SUBSEQUENT FAILURE OF SAFETY SYSTEMS	Turbine fire	TG mechanical failure, fire and flooding	Core and F
103500	REACTOR TRIP DUE TO UNDERVOLTAGE ON CONTROL ROOM INSTRUMENTATION DISTRIBUTION PANEL AT D.C. COOK UNIT 2	Transient – Loss of power to necessary plant systems	Inverter failure (silicon control rectifier)	I&C (prote
136300	REACTOR SCRAM DUE TO LOW PRESSURE SIGNAL OF THE INSTRUMENT AIR	Transient – Low IA pressure	Human error and valve failure	No
146400	INTERRUPTION OF 48V DC CLASS I POWER RESULTS IN A LOSS OF UNIT LOW PRESSURE SERVICE WATER (LPSW) AND A PARTIAL LOSS OF CLASS IV POWER	Transient - Loss of DC power to necessary systems	Failure of two converters	SWS, powe
601200	INSTRUMENT AIR FAILURE AT TAPS	Transient – Problems with CR drive mechanism	Low IA pressure (air leak due to corrosion)	No
609500	DISRUPTION OF NORMAL OPERATING CONDITIONS AND EMERGENCY SHUT-DOWN OF THE FIRST UNIT OF THE VVER REACTOR OF THE ARMENIAN NPP DUE TO FIRE	Transient – Emergency shutdown due to fire	Short circuit in the electric motor of a SW pump	Plant regula
612400	INADVERTENT ACTUATION OF SAFETY SYSTEMS RESULTING IN SCRAM	Transient – RCS pumps trip	Spurious safety signal due to an incorrect settings	No
616400	UNPLANNED SHUTDOWN OF IGNALINSKAYA 2 ON SPURIOUS SIGNALS DUE TO CABLE FIRES	Transient – Emergency shutdown due to fire	Control cable ignition due to heat-up or short circuit	CR ir signif
617400	ACTUATION OF THE FIRE FIGHTING SYSTEM AND ONE CHANNEL OF THE SAFETY SYSTEM DUE TO SPURIOUS SIGNALS	Transient – Spurious trip	Loss of power to ESFAS channel due to flooding	ESFA
6256G0	DISCONNECTION OF ALL DUKOVANY UNITS FROM GRID.	Transient – Disconnection of all units from the grid	Short circuit in a switching station (human error)	Loss EDG
6274G0	UNPLANNED SHUTDOWNS OF VVER-1000 PLANTS DUE TO DESIGN DEFICIENCIES IN SERVICE WATER SYSTEMS FOR ESSENTIAL EQUIPMENT	Transient – Spurious trip	Spurious BOP protection signal due to flooding	No
627700	ACTIVATION OF THE EMERGENCY REACTOR PROTECTION SYSTEM DUE TO SPURIOUS SIGNAL DURING DEENERGIZATION OF THE SKALA CENTRALIZED MONITORING	Transient – Spurious trip	De-energization of unit parameter monitoring (HE)	BOP
633100	BATTERY MALFUNCTION DETECTED DURING TESTING	Transient – Spurious trip	Spurious protection signals due to battery malfunction	ESFA
634400	DEFICIENCIES IN THE ORGANIZATION OF STAND-BY DIESEL AND SAFETY SYSTEM BATTERY OPERATION	Routine testing of safety systems	Battery degradation	EDG
635000	DE-ENERGIZATION OF DC SWITCHBOARD DUE TO DAMAGE TO A REVERSIBLE MOTOR GENERATOR AND BATTERY TRIPPING	Transient – Spurious trip	Loss of DC power to RPS (motor generator failure)	EDG EFW
636000	REACTOR SCRAM ON HI-HI STEAM DRUM LEVEL DUE TO DEENERGIZATION OF THE 0.4 KV BUS	Transient – Increase in feedwater flow	Loss of essential EPS bus bar (transformer failure)	TG C contr

APPENDIX C

FINAL SELECTION OF EVENTS –
CCI RELATED INSIGHTS

REPORT	EVENT TITLE	EVENT	ELEMENTS INVOLVED IN DIRECT CAUSE		
			Human Action	Failure/fault description	Component/ System type
000200	INADVERTENT CLOSURE OF ALL MAIN STEAM ISOLATION VALVES	P-CCI	TE*	Valve closure	M/M
001104	PARTIAL FAILURE OF THE SCRAM SYSTEM	CCF	-	Valve leakage	M/I
004002	LOSS OF 125V DC BUS.	CCI	OE*	Breaker opening	E/E
006200	SPURIOUS SAFETY INJECTION DUE TO HIGH DIFFERENTIAL PRESSURE BETWEEN STEAM LOOPS.	P-CCI	-	Valve stack	M/I
009702	FAILURE OF HIGH PRESSURE SAFETY INJECTION SYSTEM	P-CCI	-	Power supply fail	E/E
012500	PARTIAL FAILURE OF THE THREE PHASE SUPPLY SYSTEM.	CCI	TE*	Converter disconnected	E/E
014800	FIRE RESULTING FROM TRANSFORMER FAILURE.	CCI	-	Transformer failure	E/E
016500	REACTOR SCRAM AND LOSS OF REDUNDANT SAFETY SIGNALS.	P-CCI	-	Transmitter valve open	M/I
0176G4	INADVERTENT ACTUATION OF FIRE SUPPRESSION SYSTEM.	P-CCI	TA	Fire spray actuated	I/I
021600	LOSS OF REGULATION INCIDENT.	PDM	-	Computer HW	I/I
0218G2	STEAM EROSION IN TURBINE EXHAUST LINES	PDM	-	Pipe break	M/M
0218G4	EROSION AND RUPTURE OF HEATER DRAIN PIPING, PARTLY DUE TO MISUSE OF PUMP DISCHARGE PIPE	PDM	-	Pipe break	M/M
0218G5	FEEDWATER LINE RUPTURE DUE TO EROSION IN AREA NOT REGULARLY INSPECTED	PDM	-	Pipe break	M/M
022100	BLACKOUT AFFECTING THE DOEL POWER STATION	CCI	-	Spurious signal	I/I
022900	LEAKAGE IN THE CHEMICAL AND VOLUME CONTROL SYSTEM	PDM	-	Defective gasket	M/M
0241G1	FROZEN INSTRUMENTATION LINES CAUSE INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES.	PDM	-	Frozen instrument line	M/I
025800	LOSS OF AUXILIARY POWER AFFECTS TWO UNITS.	PDM	ME*	Fuses pulled out	E/E
027002	DRAIN OF PRIMARY WATER DUE TO MISALIGNMENT OF A RHR VALVE	P-CCI	ME*	Power supply connected	E/E
029600	REACTOR SCRAM DUE TO EXCESSIVE VALVE LEAKAGE	PDM	-	Valve leakage	M/M
030302	TROUBLE WITH ELECTRICAL SUPPLY SYSTEM CAUSED BY LIGHTNING	CCI	-	Melted fuse	E/I
031900	IMPACT OF A LIGHTNING STROKE INTO PLANT'S 220 KV LINE	PDM	-	Electronic device failed	I/I
032700	MAIN FEEDWATER LINE BREAK DUE TO WATER HAMMER.	PDM	-	Spurious signal	I/I
037500	UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT	P-CCI	-	Valve closure	M/M
042503	VOLTAGE DROP FOLLOWED BY LOSS OF TRAIN A 48 VOLT BUS LEADING TO LOSS OF BOTH OFF-SITE POWER AND TRAIN A DIESEL GENERATOR	CCI	HE+	Electronic card failure	I/I
0437G5	TESTING RESULTS IN STATION BLACKOUT	CCI	TE*	Electric switch open	E/E
044300	WATER HAMMER IN FEEDWATER PIPING AND SUBSEQUENT SCRAM DUE TO FEEDWATER SYSTEM PROBLEMS	P-CCI	OE+	Check valve failure	M/M
049500	HIGH PRESSURE COOLANT INJECTION SYSTEM LOCKOUT.	PDM	TE+	Valve failure	M/M
052300	EXPLOSION AND FIRE IN AUXILIARY TRANSFORMER RESULTS IN LOSS OF STARTUP TRANSFORMER	CCI	-	Transformer failure	E/E
053900	SWITCHYARD COMPUTER DESIGN DEFICIENCY CAUSES LOSS OF NORMAL OFF-SITE POWER.	PDM	ME+	Breaker failure	E/E
057000	INOPERABLE SAFETY INJECTION PUMPS	CCF	PE+	Piping blockage	M/M
058803	LOSS OF ALL IN-PLANT AC POWER, REACTOR TRIP AND WATER HAMMER AT SAN ONOFRE-1	CCI	-	Transformer failure	E/E
059400	SYSTEM INTERACTION EVENT RESULTING IN REACTOR SYSTEM SAFETY RELIEF VALVE OPENING FOLLOWING A FIRE PROTECTION DELUGE SYSTEM MALFUNCTION.	CCI	OP+	Valve failure	M/I
061000	FLOODING RESULTS FROM EXPANSION JOINT FAILURE AND INSTALLATION ERROR	PDM	-	Expansion joint failure	M/M
061600	SAFETY INJECTION AND REACTOR TRIP DUE TO LOSS OF STATION INSTRUMENT AIR PRESSURE.	CCI	-	Faulty solder/fitting	M/I
061800	BLACKOUT SIGNAL AND INTERACTION EVENT BETWEEN UNITS	CCI	TE*	Breaker opened	E/E
063300	REACTOR TRIP RESULTS FROM ERRONEOUS CONTROL BOARD INFORMATION DUE TO INVERTER FAILURE.	CCI	-	Inverter failure	E/E
064100	INADVERTENT OPENING OF A RELIEF VALVE DUE TO A SPURIOUS CONTACT BETWEEN WIRES INSIDE AN ELECTRIC PENETRATION	PDM	-	Wire short circuit	E/E
064700	INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION AND SUBSEQUENT REACTOR TRIP	CCI	-	Ventilation fan failure	M/E
064800	LOSS OF INSTRUMENT AIR CAUSES REACTOR SCRAM WITH HIGH PRESSURE COOLANT INJECTION SUBSEQUENTLY FAILING.	CCI	MA	Instrument air dryer	M/I
067800	WATER DRIPPING IN CONTROL ROOM.	P-CCI	-	Valve open	M/M
069300	UNLABELED SWITCH RESULTS IN INADVERTENT ACTUATION OF DELUGE SPRAY SYSTEM AND SUBSEQUENT SCRAM	P-CCI	ME*	Wrong switch	I/I

REPORT	EVENT TITLE	EVENT	ELEMENTS INVOLVED IN DIRECT CAUSE		
			Human Action	**Failure/fault description**	**Component/ System type**
086100	FALSE TRIGGERING OF REACTOR PROTECTION SIGNALS DUE TO A FAILED CLOCK GENERATOR	P-CCI	-	Clock gen. short circuit	I/I
091400	FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS	CCI	-	Pipe flow blockage	M/M
095800	PARTIAL BLOCKAGE OF THE WATER INTAKE OF ONE UNIT, AND LOSS OF OFF-SITE POWER TO THE TWIN DURING COLD WEATHER	PDM	-	Water intake blockage	M/M
102502	FIRE IN ONE TURBINE GENERATOR GROUP AND SUBSEQUENT FAILURE OF SAFETY SYSTEMS	CCI	-	Turbine/generator failure	M/M
103500	REACTOR TRIP DUE TO UNDERVOLTAGE ON CONTROL ROOM INSTRUMENTATION DISTRIBUTION PANEL AT D.C. COOK UNIT 2	CCI	-	Rectifier failure	E/E
136300	REACTOR SCRAM DUE TO LOW PRESSURE SIGNAL OF THE INSTRUMENT AIR	PDM	OE+	Valve failure	M/M
146400	INTERRUPTION OF 48V DC CLASS I POWER RESULTS IN A LOSS OF UNIT LOW PRESSURE SERVICE WATER (LPSW) AND A PARTIAL LOSS OF CLASS IV POWER	CCI	-	Converter failure	E/E
601200	INSTRUMENT AIR FAILURE AT TAPS	P-CCI	-	Instrument air line leak	M/I
609500	DISRUPTION OF NORMAL OPERATING CONDITIONS AND EMERGENCY SHUT-DOWN OF THE FIRST UNIT OF THE VVER REACTOR OF THE ARMENIAN NPP DUE TO FIRE	CCI	-	Short circuit	E/E
612400	INADVERTENT ACTUATION OF SAFETY SYSTEMS RESULTING IN SCRAM	PDM	TE*	El. Bus disconnection	E/E
616400	UNPLANNED SHUTDOWN OF IGNALINSKAYA 2 ON SPURIOUS SIGNALS DUE TO CABLE FIRES	CCI	-	Cable overheating	E/E
617400	ACTUATION OF THE FIRE FIGHTING SYSTEM AND ONE CHANNEL OF THE SAFETY SYSTEM DUE TO SPURIOUS SIGNALS	CCI	-	Spurious signal	I/I
6256G0	DISCONNECTION OF ALL DUKOVANY UNITS FROM GRID.	PDM	OE	Switchyard short circuit	E/E
6274G0	UNPLANNED SHUTDOWNS OF VVER-1000 PLANTS DUE TO DESIGN DEFICIENCIES IN SERVICE WATER SYSTEMS FOR ESSENTIAL EQUIPMENT	PDM	-	Tank overfilling	M/M
627700	ACTIVATION OF THE EMERGENCY REACTOR PROTECTION SYSTEM DUE TO SPURIOUS SIGNAL DURING DEENERGIZATION OF THE SKALA CENTRALIZED MONITORING	PDM	OE*	Monitoring unit de-energ.	I/E
633100	BATTERY MALFUNCTION DETECTED DURING TESTING	P-CCI	OE	Battery low voltage	E/E
634400	DEFICIENCIES IN THE ORGANIZATION OF STAND-BY DIESEL AND SAFETY SYSTEM BATTERY OPERATION	PDM	-	Battery low voltage	E/E
635000	DE-ENERGIZATION OF DC SWITCHBOARD DUE TO DAMAGE TO A REVERSIBLE MOTOR GENERATOR AND BATTERY TRIPPING	CCI	-	Motor-generator failure	E/E
636000	REACTOR SCRAM ON HI-HI STEAM DRUM LEVEL DUE TO DEENERGIZATION OF THE 0.4 KV BUS	PDM	-	Transformer failure	E/E

ABBREVIATIONS USED

Human actions

TA	testing activities
MA	maintenance activities
TE	testing error
ME	maintenance error
OE	operational error
*	human error / no hardware failures involved
+	human error / additional hardware failures involved

Failures/faults

M	Mechanical component /system
E	Electrical component / system
I	Instrumentation & control component / system

APPENDIX D

IE ASSIGNMENT TO EPRI IE LIST

REPORT	EVENT TITLE	EVENT OF	IE TYPE
000200	INADVERTENT CLOSURE OF ALL MAIN STEAM ISOLATION VALVES	P-CCI	Transient - Clo
001104	PARTIAL FAILURE OF THE SCRAM SYSTEM	CCF	Transient - Ma
004002	LOSS OF 125V DC BUS.	CCI	Transient - Los
006200	SPURIOUS SAFETY INJECTION DUE TO HIGH DIFFERENTIAL PRESSURE BETWEEN STEAM LOOPS.	P-CCI	Transient - Ina
009702	FAILURE OF HIGH PRESSURE SAFETY INJECTION SYSTEM	P-CCI	Transient - Los
012500	PARTIAL FAILURE OF THE THREE PHASE SUPPLY SYSTEM.	CCI	Transient - Lo
014800	FIRE RESULTING FROM TRANSFORMER FAILURE.	CCI	Transient - Lo
016500	REACTOR SCRAM AND LOSS OF REDUNDANT SAFETY SIGNALS.	P-CCI	Transient – ina
0176G4	INADVERTENT ACTUATION OF FIRE SUPPRESSION SYSTEM.	P-CCI	Transient – m
021600	LOSS OF REGULATION INCIDENT.	PDM	Transient – Inc
0218G2	STEAM EROSION IN TURBINE EXHAUST LINES	PDM	Transient – Ma
0218G4	EROSION AND RUPTURE OF HEATER DRAIN PIPING, PARTLY DUE TO MISUSE OF PUMP DISCHARGE PIPE	PDM	Transient – Tu
0218G5	FEEDWATER LINE RUPTURE DUE TO EROSION IN AREA NOT REGULARLY INSPECTED	PDM	Transient – Ma
022100	BLACKOUT AFFECTING THE DOEL POWER STATION	CCI	Transient – Lo
022900	LEAKAGE IN THE CHEMICAL AND VOLUME CONTROL SYSTEM	PDM	Transient – Ma
0241G1	FROZEN INSTRUMENTATION LINES CAUSE INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES.	PDM	Transient – ES
025800	LOSS OF AUXILIARY POWER AFFECTS TWO UNITS.	PDM	Transient – Lo
027002	DRAIN OF PRIMARY WATER DUE TO MISALIGNMENT OF A RHR VALVE	P-CCI	Transient – Int
029600	REACTOR SCRAM DUE TO EXCESSIVE VALVE LEAKAGE	PDM	Transient – Ma
030302	TROUBLE WITH ELECTRICAL SUPPLY SYSTEM CAUSED BY LIGHTNING	CCI	Transient – Inc
031900	IMPACT OF A LIGHTNING STROKE INTO PLANT'S 220 KV LINE	PDM	Transient – Ge
032700	MAIN FEEDWATER LINE BREAK DUE TO WATER HAMMER.	PDM	Transient – Lo
037500	UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT	P-CCI	Transient – Cl
042503	VOLTAGE DROP FOLLOWED BY LOSS OF TRAIN A 48 VOLT BUS LEADING TO LOSS OF BOTH OFF-SITE POWER AND TRAIN A DIESEL GENERATOR	CCI	Transient - Los
0437G5	TESTING RESULTS IN STATION BLACKOUT	CCI	Transient - Los
044300	WATER HAMMER IN FEEDWATER PIPING AND SUBSEQUENT SCRAM DUE TO FEEDWATER SYSTEM PROBLEMS	P-CCI	Transient – fee
049500	HIGH PRESSURE COOLANT INJECTION SYSTEM LOCKOUT.	PDM	Transient – Mi
052300	EXPLOSION AND FIRE IN AUXILIARY TRANSFORMER RESULTS IN LOSS OF STARTUP TRANSFORMER	CCI	Transient – Fa
053900	SWITCHYARD COMPUTER DESIGN DEFICIENCY CAUSES LOSS OF NORMAL OFF-SITE POWER.	PDM	Transient – tra
057000	INOPERABLE SAFETY INJECTION PUMPS	CCF	SI pumps foun
058803	LOSS OF ALL IN-PLANT AC POWER, REACTOR TRIP AND WATER HAMMER AT SAN ONOFRE-1	CCI	Transient – Ma
059400	SYSTEM INTERACTION EVENT RESULTING IN REACTOR SYSTEM SAFETY RELIEF VALVE OPENING FOLLOWING A FIRE PROTECTION DELUGE SYSTEM MALFUNCTION.	CCI	Transient – Ma
061000	FLOODING RESULTS FROM EXPANSION JOINT FAILURE AND INSTALLATION ERROR	PDM	Transient – Lo
061600	SAFETY INJECTION AND REACTOR TRIP DUE TO LOSS OF STATION INSTRUMENT AIR PRESSURE.	CCI	Transient – Mi
061800	BLACKOUT SIGNAL AND INTERACTION EVENT BETWEEN UNITS	CCI	Transient – Lo
063300	REACTOR TRIP RESULTS FROM ERRONEOUS CONTROL BOARD INFORMATION DUE TO INVERTER FAILURE.	CCI	Transient – Ma

REPORT	EVENT TITLE	EVENT OF	IE TYPE
064100	INADVERTENT OPENING OF A RELIEF VALVE DUE TO A SPURIOUS CONTACT BETWEEN WIRES INSIDE AN ELECTRIC PENETRATION	PDM	Transient – Ina
064700	INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION AND SUBSEQUENT REACTOR TRIP	CCI	Transient – SC
064800	LOSS OF INSTRUMENT AIR CAUSES REACTOR SCRAM WITH HIGH PRESSURE COOLANT INJECTION SUBSEQUENTLY FAILING.	CCI	Transient – Inc
067800	WATER DRIPPING IN CONTROL ROOM.	P-CCI	Transient – Ma
069300	UNLABELED SWITCH RESULTS IN INADVERTENT ACTUATION OF DELUGE SPRAY SYSTEM AND SUBSEQUENT SCRAM	P-CCI	Transient – Inc
086100	FALSE TRIGGERING OF REACTOR PROTECTION SIGNALS DUE TO A FAILED CLOCK GENERATOR	P-CCI	Transient – Fa
091400	FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS	CCI	Transient – Ins
095800	PARTIAL BLOCKAGE OF THE WATER INTAKE OF ONE UNIT, AND LOSS OF OFF-SITE POWER TO THE TWIN DURING COLD WEATHER	PDM	Transient – Lo
102502	FIRE IN ONE TURBINE GENERATOR GROUP AND SUBSEQUENT FAILURE OF SAFETY SYSTEMS	CCI	Turbine fire
103500	REACTOR TRIP DUE TO UNDERVOLTAGE ON CONTROL ROOM INSTRUMENTATION DISTRIBUTION PANEL AT D.C. COOK UNIT 2	CCI	Transient – Lo
136300	REACTOR SCRAM DUE TO LOW PRESSURE SIGNAL OF THE INSTRUMENT AIR	PDM	Transient – Lo
146400	INTERRUPTION OF 48V DC CLASS I POWER RESULTS IN A LOSS OF UNIT LOW PRESSURE SERVICE WATER (LPSW) AND A PARTIAL LOSS OF CLASS IV POWER	CCI	Transient - Los
601200	INSTRUMENT AIR FAILURE AT TAPS	P-CCI	Transient – Pro
609500	DISRUPTION OF NORMAL OPERATING CONDITIONS AND EMERGENCY SHUT-DOWN OF THE FIRST UNIT OF THE VVER REACTOR OF THE ARMENIAN NPP DUE TO FIRE	CCI	Transient – En
612400	INADVERTENT ACTUATION OF SAFETY SYSTEMS RESULTING IN SCRAM	PDM	Transient – RC
616400	UNPLANNED SHUTDOWN OF IGNALINSKAYA 2 ON SPURIOUS SIGNALS DUE TO CABLE FIRES	CCI	Transient – En
617400	ACTUATION OF THE FIRE FIGHTING SYSTEM AND ONE CHANNEL OF THE SAFETY SYSTEM DUE TO SPURIOUS SIGNALS	CCI	Transient – Sp
6256G0	DISCONNECTION OF ALL DUKOVANY UNITS FROM GRID.	PDM	Transient – Di
6274G0	UNPLANNED SHUTDOWNS OF VVER-1000 PLANTS DUE TO DESIGN DEFICIENCIES IN SERVICE WATER SYSTEMS FOR ESSENTIAL EQUIPMENT	PDM	Transient – Sp
627700	ACTIVATION OF THE EMERGENCY REACTOR PROTECTION SYSTEM DUE TO SPURIOUS SIGNAL DURING DEENERGIZATION OF THE SKALA CENTRALIZED MONITORING	PDM	Transient – Sp
633100	BATTERY MALFUNCTION DETECTED DURING TESTING	P-CCI	Transient – Sp
634400	DEFICIENCIES IN THE ORGANIZATION OF STAND-BY DIESEL AND SAFETY SYSTEM BATTERY OPERATION	PDM	Routine testing
635000	DE-ENERGIZATION OF DC SWITCHBOARD DUE TO DAMAGE TO A REVERSIBLE MOTOR GENERATOR AND BATTERY TRIPPING	CCI	Transient – Sp
636000	REACTOR SCRAM ON HI-HI STEAM DRUM LEVEL DUE TO DEENERGIZATION OF THE 0.4 KV BUS	PDM	Transient – Inc

APPENDIX E

EXAMPLE ANALYSIS OF SELECTED EVENTS

EVENT ANALYSIS SUMMARY		
Event description	Event code	**022100**
	Title	**BLACKOUT AFFECTING THE DOEL POWER STATION**
Direct cause	Description	Test of TG protection system at Unit 3
	Type	Spurious signal
	Equipment Involved	TG protection system

Relevant	Occurrence/ Plant status	Occurrence description/ System involved and their degradation	Behaviour	Consequence of
	OC1	Spurious protection signal of the turbine	ABN	F1
	OC2	Unit 3 Turbine trip	CR	OC1
	OC3	Unit 3 Reactor trip	CR	OC2
	OC4	380 kV grid collapse	CR	OC3
	OC5	Fast switch-over of the load to 150 kV grid failed; grid in degraded status for ½ hour	ABN	F2
	OC6	Main circuit breakers of Units 1 & 2 trip due to grid instability	CR	OC4
	OC7	Unit 1 Reactor trip (on low frequency) Unit 2 Reactor trip (on high flux)	CR	OC6
	OC8	DGs (common to Units 1 & 2) failed	ABN	F3

Event Classification	Accident sequence Initiated	Loss of all off-site power
	EPRI IE	35
	Degraded Mitigation Systems	All off-site power supply sources, on-site AC power
	Dependencies Involved	Disturbances in the grid caused by one of the 3 Units (failure F1); other failures in electrical systems (failures F2 and F3) were independent
	Group of interest	CCI (limited extent – some failures were independent)
	Significance to safety	High – demonstrated strong dependencies in electrical system

Note: Abbreviations used

OCi Occurrence No.i
ABN Abnormal behaviour
CR Consequential behaviour
HEi Human error No. i
Fi Failure No. i

EVENT ANALYSIS SUMMARY		
Event identification	Event code	**027002**
	Title	**DRAIN OF PRIMARY WATER DUE TO MISALIGNMENT OF A RHR VALVE**
Direct cause	Description	Valve switch left on the „open„ position during a maintenance, power restored mistakenly during a test for overall containment leakage
	Type	Human errors/ energising a misaligned valve
	Equipment involved	RHR valve control logic and valve power supply

Relevant	Occurrence	Occurrence description/ System involved and their degradation	Behaviour	Consequence of
	OC1	RHR - A valve switch in the CR left on the „open„ position during a maintenance	ABN	HE1
	OC2	RHR - Maintenance operator mistakenly energises a misaligned valve	ABN	HE2
	OC3	RCS - Drain of the primary coolant through an open valve results in a low reactor coolant level	CR	OC1$\wedge$OC2
	OC4	RPS - scram on „low level„ signal	CR	OC3
	OC5	RHR - isolated automatically	CR	OC3

Event Classification	Accident sequence Initiated	Intermediate System LOCA
	EPRI IE	No
	Degraded Mitigation Systems	RHR
	Dependencies Involved	Direct dependency between IE and system degradation; Human errors HE1 and HE2 (independent) $\Rightarrow$ loss of coolant $\Rightarrow$ RHR degradation (isolation)
	Group of interest	Potential CCI
	Significance to safety	Medium – Alternative mitigation measures not affected

Note: Abbreviations used

OCi Occurrence No.i
ABN Abnormal behaviour
CR Consequential behaviour
HEi Human error No. i
Fi Failure No. i

EVENT ANALYSIS SUMMARY		
Event identification	Event code	**042503**
	Title	**VOLTAGE DROP FOLLOWED BY LOSS OF TRAIN A 48 V BUS LEADING TO LOSS OF BOTH OFF-SITE POWER AND TRAIN A DIESEL GENERATOR**
Direct cause	Description	Control card failure led to a failure of rectifier followed by battery discharge; a low voltage CR alarm not noticed
	Type	I&C failure not restored due to human error
	Equipment Involved	Rectifier in the DC power supply system

Relevant occurrences	Occurrences/ Plant status	Occurrence description/ System involved and their degradation	Behaviour	Consequence of
	OC1	Control card failure in DC train A rectifier followed by the buffer battery discharge	ABN	F1
	OC2	Low voltage CR alarm not noticed, no action to restore DC voltage	ABN	HE1
	OC3	Low DC voltage	ABN	F1∧HE1
	Plant status	Degradation of the RPS and the EPS logic circuitry due to a low DC voltage	CR	OC3
	OC4	Reactor trip	CR	OC3
	OC5	Loss of off-site power	CR	OC4∧OC3
	OC6	Train A DG fails to start	CR	OC3
	OC7	train B DG starts and feeds the essential systems	CR	OC6
	OC8	220 V fuse blown due to an electrical transient	CR	OC5∧OC6
	Plant status	Loss of information in the CR	CR	OC8
	OC9	RCS pumps trip and PRZ discharge valve open due to RCS pressure transient	CR	OC5∧OC4

Event Classification	Accident sequence Initiated	Loss of off-site power
	EPRI IE	No
	Degraded Mitigation Systems	Emergency DG, CR indications
	Dependencies Involved	Direct dependencies through the DC system; low voltage ⇒ degradation of RPS and EPS control ⇒ loss of off-site power ⇒ further degradation of plant systems (CR instrumentation)
	Group of interest	CCI
	Significance to safety	High – because of the extent and importance of plant system degradation

Note: Abbreviations used

OCi Occurrence No.i; ABN Abnormal behaviour; CR Consequential behaviour;
HEi Human error No. I;

Fi	Failure No. i

EVENT ANALYSIS SUMMARY		
Event identification	Event code	**0437G5**
	Title	**TESTING RESULTS IN STATION BLACKOUT**
Direct cause	Description	HEs during a plant test „Loss of TG and off-site power„
	Type	Inadvertent human action, ineffective verification
	Equipment Involved	Electrical distribution system

Relevant	Occurrence/ Plant status	Occurrence description/ System involved and their degradation	Behaviour	Consequence of
	OC1	Wrong DC supply knife switches opened in electrical distribution system	ABN	HE1
	OC2	Emergency Safety System (ESS) logic circuitry – loss of DC power supply	CR	OC1
	Plant Status	Emergency Power Supply degraded - No automatic transfer of ESS buses to alternate source - No automatic DG startup unavailable - Inability to re-energise from off-site - No automatic load shedding CR – most instrumentation failed LP ECCS- unavailable	CR	OC2
	OC3	First mistake not noticed – test continued, Off-site power disconnected		HE2
	OC4	Plant AC power – totally lost	CR	OC3$\wedge$OC2

Event Classification	Accident sequence Initiated	Loss of AC power
	EPRI IE	35, 37
	Degraded Mitigation Systems	EPS, CR indications, ECCS
	Dependencies Involved	Direct dependency between IE and system degradation; Human errors HE1 and HE2 dependent (test related) HE1 $\Rightarrow$ degradation of EPS, CR indications, ECCS HE2 $\Rightarrow$ IE (Loss of AC power)
	Group of interest	CCI
	Significance to safety	High – Significant degradation of important mitigation systems

Note: Abbreviations used

OCi	Occurrence No.i
ABN	Abnormal behaviour
CR	Consequential behaviour
HEi	Human error No. i
Fi	Failure No. i

APPENDIX F

EVENTS SELECTED FOR DETAILED ANALYSIS – EVENT DESCRIPTION

Report # 000200 INADVERTENT CLOSURE OF ALL MAIN STEAM ISOLATION VALVES

An operator error, during a routine function-test of the pilot valves on a medium operated isolation valve in one of the main steam lines of the turbine, led to inadvertent closure of this valve. The increased steam pressure caused the medium operated safety valve in this loop to open and close several times. In order to close the safety valve the main coolant pump was stopped manually and the power was reduced. The following faster steam line pressure gradient led to isolation of all steam lines and the reactor and the turbine tripped automatically. Some time after the main-steam line isolation, the main condensate pumps were tripped as a consequence of an independent failure in the condensor hot-well level control. A backflow of steam occurred through the pre-heated condensate line by way of the minimum flow piping of the condensate pumps. Steam condensation banging then ensued in the condensate line. The resulting piping oscillations caused damage to some pipe and component supports, and several valve position motors broke off. Following Actions were taken: - relapping of pilot valves and setpoints for testing; - investigation of the optimisation of the valve function in particular on closing behaviour. Tentatively, the test will be carried out at reduced power operation; - the condensate pump minimum flow line will be automatically isolated following a trip of all main condensate pumps.

Report # 000300 LOSS OF REACTOR COOLANT FROM THE HIGH PRESSURE REACTOR COOLANT SYSTEM

A technician working on an I&C board which had misaligned connector pins caused the feedwater control valve to reduce flow. The pressure and temperature of the primary coolant rose as one steam generator went dry. The relief valve opened and the reactor was shutdown automatically. The coolant pressure dropped and coolant discharged into the drain tank and then 40,000 gallons of coolant water were dumped into the reactor containment after the tank rupture disk actuated. High pressure injection pumps actuated automatically. Further, the operator turned off the coolant pumps to recover pressure and shut a block valve to stop leakage. The following actions were recommended. Training of Operators, Improved procedures, Design improvement.

Report # 001104 PARTIAL FAILURE OF THE SCRAM SYSTEM

As part of the normal scheduled shutdown routine power had been reduced and a manual scram initiated. Of the 185 control rods, 75 failed to fully insert. 4 scram operations and about 15 minutes were required in order to get all rods inserted. The cause was water accumulation in the scram discharge volume prior to the attempted scram. Inadequate scram volume to which to vent water prevents full rod insertion. Instrumentation deficiencies were identified for detecting water in the scram discharge volume. Degraded ir pressure to close valves was shown to be a possible cause of valve leakage to the scram volume. Recommended actions taken were - The operability of scram function should be independent of the scram discharge volume venting and draining requirements - Scram volume water level monitoring instruments should be redundant and diverse - All related vent and drain paths should have redundant automatic isolation valves - Emergency operating procedures and operator training should be provided for complete and partial scram failure conditions - Modification of the related vent and drain arrangement The follow up reports are 11.02, 11.03 and 11.04. IRS11.02 reports a precursor of this incident.

Report # 0014G5 PIPE RUPTURE IN THE REACTOR WATER CLEAN-UP SYSTEM

A pipe (150 mm diameter) of the reactor water clean-up system ruptured and water of 5 cubic meters was discharged in the room, followed by the isolation of the reactor building isolation monitoring system, the reactor trip, the isolation of the clean-up system and the turbine trip. The crack in the pipe was 150 mm in length and 2 mm in width. The crack was caused by the pipe tension induced by the mixing of two flows with different temperature at T-joint of the pipe line. Two flows with different temperature were caused by an operating error due to an inaccurate operating instruction. In addition to the pipe repair, the operating instruction was revised. See IRS0016.G8 for similar incident in xxxx. [This incident is described in IRS14.]

Report # 0014G8 RAPID TEMPERATURE DECREASE OF REACTOR PRESSURE VESSEL DUE TO OPERATING ERROR

Due to problems with the generator rotor grounding, the reactor was brought to a hot standby from 20% power. MSIV was closed to cut off the steam supply to turbine condenser. Due to inadequate instructions, the discharge valve to control system pressure was left under manual operation and totally open. When the fast-opening valve located before the discharge valve was opened, the intended control function was not achieved, i.e., the valve continued to blow. As a consequence of the blow down, the water level in the reactor dropped, hence the speed of feed water pumps increased automatically. The additional cold water caused a flux increase followed by the reactor trip. The pressure drop and the open discharge valve were detected and the valve was closed. The pressure drop was equivalent to a temperature change of 68 C/12 min., while the technical specification limit was 7,5 C/10 min.. However, the temperature drop was approximately 10 C greater than the analyzed event calculated to occur 10 times over the entire service life of the reactor. As counter measures, the instructions on the relief system were modified and the requirements for reactor pressure control were added. The similar incident at xxxx was described in IRS0014G5. [This incident is described in IRS14.]

Report # 0019G5 INHIBITION OF SAFETY INJECTION AFTER SPONTANEOUS OPENING OF THE PRESSURISER SPRAY REGULATION VALVE

A mechanical fault caused a spurious opening of the pressurizer spray regulating valve, causing a rapid fall in primary pressure and emergency shut down as a result of low pressurizer pressure. The operator prevented safety injection violating technical specification, having identified the fault with the help of the display panel and keeping continuous watch on the sub-cooling recorder. As remedial actions, directives were issued to remind operators that they are not allowed to inhibit ECCS signals, regardless of their interpretation of the incident. [This incident is compiled in IRS19]

Report # 004002 LOSS OF 125V DC BUS.

A plant equipment operator inadvertently opened a 125 volts DC main feeder breaker causing the loss of one of the two redundant DC emergency systems which led to a reactor trip. The loss of this DC system prevented the main turbine from tripping automatically as designed and it was manually tripped 30 seconds later. The trip of the turbine coupled with the inoperability of this DC system caused the loss of offsite power to one of the two redundant alternating current (AC) systems and the automatic starting of both redundant diesel generators. Subsequently, both of the diesel generators tripped automatically as a result of an inherent design trip feature in the control circuits of one of the diesel generators and a mechanical failure in the other. All annunciators were lost and several indications were not available (including auxiliary feedwater flow to a SG, main steam header pressure, make-up tank level, charging pump pressure, etc). Also pressurizer spray was not available through the normal spray system. One to two hours later, all fuses were replaced and the instrumentation restored. As Actions Taken, the licensee provided permanent breaker identification labels and reviewed the plant equipment operators rounds to identify similar situations. [This is a follow-up of IRS 40].

Report # 004100 xxxx REACTOR SITE ALERT DUE TO HIGH LEAKAGE INTO DRYWELL

At 12.35 a.m. a leak of an unidentified origin in the drywell occurred. The maximum leak rate was 21.5 gallons per minute (GPM). The reactor was operating at 100 percent power at the time. The reactor was manually scrammed from 30 percent power at 1.23 a.m. There was no radioactive release to the environment and no contamination of personnel. The licensee declared a site alert at 12.35 a.m., due to the potential to exceed the leak rate of 25 GPM. The site alert was terminated at 2.24 a.m. Preparations were then being made to determine the source of leakage. The plant was taken to cold shutdown later on May 22 and that night workers determined the leak was caused by the loss of packing material from one of the recirculation discharge valves. The valve was repacked and the plant was returned to power at approximately 17.00 on May 25.

Report # 006200 SPURIOUS SAFETY INJECTION DUE TO HIGH DIFFERENTIAL PRESSURE BETWEEN STEAM LOOPS.

During nuclear tests before initial criticality, the safety injection signal was activated due to high pressure difference between two steam generators. Pressure was being increased at the time. The cause of the safety injection signal was found to be that two isolating valves of the pressure difference transmitter had their valves stuck to the seat even though apparently open. They were on the side of the transmitter which did not see the rising pressure.

Report # 006600 INADVERTENT SAFETY INJECTION.

xxxx was operating at 837 MWe when a disturbance on one of the connections to the 400 kV grid was obtained. The corresponding generator protection system (xxx has two turbine generator sets) tripped incorrectly and thus the generator was disconnected from the grid and switched over to house load operation. At the switch over, however, the reactor was tripped due to a too high negative flux rate. Eight minutes after these initial events safety injection was obtained due to faulty signals from the protection system caused by electrical disturbance in an inverter. Corrective measures to re-establish the voltage feed were immediately taken. At this moment a micro-switch (a fuse device) tripped with the consequence that the safety injection system could not be canceled from the control room. Meanwhile the reactor coolant system level increased and a short pressuriser relief valve opening to the pressuriser relief tank took place. The tripped micro-switch (fuse device) was discovered then reset and conditions could return to normal. This chain of events was caused by several independent imperfections. Investigations and subsequent tests resulted in exchange of faulty inverter components and additional instructions applicable to the inverter were issued. Furthermore, the inverter alarm signals were modified at the outage in 1980.

Report # 007200 HIGH PRESSURE COOLANT INJECTION SYSTEM'S FAILURE TO AUTOMATICALLY START FOLLOWING REACTOR TRIP

The reactor scrammed due to a main turbine stop valve fast closure following a turbine trip. The HPCI system received an auto initiation signal on low reactor water level but failed to inject to the RPV due to isolating on a steam line high dP isolation signal. The isolation was reset and when the inboard isolation valve was opened, the outboard isolation valve received another steam line high dP isolation signal. The isolation was reset, the isolation valves opened and HPCI auto started and injected to the RPV and was used to control water level. ADS, Core Spray and LPCI systems were operable. The RCIC system was inoperable. The isolation due to steam line high dP, was determined to be caused by poor turbine speed control due to the turbine control system being out of dynamic calibration. Calibration, surveillance and operability procedures are being revised to include quickstart test for both Unit 1 and Unit 2 HPCI and RCIC systems. In addition, performance testing procedures to monitor HPCI and RCIC system performance are being developed and will be utilised to assure continued satisfactory system dynamic response. Cold quickstarts of the Unit 1 HPCI and RCIC systems are scheduled to be performed prior to placing the generator back on line.

Report # 007800 ACTUATION OF SAFETY INJECTION DUE TO ADJUSTMENT FAILURE OF STEAM FLOW TRANSMITTERS.

Shortly after coupling to the grid there was a rapid load increase which was observed and action was taken to reduce it. The rapid load increase had caused a fall of average temperature of the primary coolant. Load reduction had begun when the reactor tripped and safety injection commenced. The initiating signal was derived from low average temperature plus high steam flow. Reactor and turbine tripped, safety injection started, steam and feedwater isolation occurred together with phase "A" containment isolation. The low average temperature signal was genuine. The high steam flow signal was spurious due to a combination of peaks due to oscillations and out of adjustment of the steam flow meters. The oscillations were caused by instability of the turbine control valve electro hydraulic system. During the incident one RHR pump failed to start due to loose connections in the output relays of the solid state protection system. In order to prevent a reactor tripping incident or the actuation of the safety injection system following connection of the generator to the network, several operating rules were improved.

Report # 009702 FAILURE OF HIGH PRESSURE SAFETY INJECTION SYSTEM

At about 3:30 a.m. one of the regulated power supplies serving one of the two redundant paths of the

Reactor Protection System and a portion of the Control and Indication System failed. As a result, feedwater and steam flow and steam generator water level indications were lost for the steam generator served by this power supply and oscillations were observed in similar flow and level indications of the other two steam generators. The operators placed the steam generator water level controls under manual control and then manually tripped the reactor. During this period, high steam generator water levels resulted in increased cooling of the reactor coolant system (RCS), thereby reducing the system pressure. While SIAS was present valves in both lines from safety injection to the feedwater line failed. The valves did open when the feedwater pumps were tripped and a safety injection actuation signal (SIAS) was automatically initiated. Subsequently, RCS pressure increased, SIAS was reset at 4:00 a.m., and safety injection terminated. The sequence for safety injection was altered, valve bonnet leak-off lines were installed, and detailed procedures for testing valve and system modifications were prepared.

Report # 011800 MANUAL STOP DUE TO TROUBLE IN THE MAIN FEEDWATER CONTROL VALVE IN THE REACTOR FEEDWATER SYSTEM

Shortly after starting up after annual inspection the reactor was operating at reduced power and fluctuations were observed in feedwater flow in one of the steam generators. The related parameters were kept under observation. After that the feedwater control valve was not responding correctly to its control signal and the level in the steam drum was observed to be rising gradually. The reactor was manually tripped. The problem was traced to the feedwater control valve sticking. Investigations on this flow control valve revealed that rubbing between the valve actuate rod and the rod guide bushing made the rod driving awkward. As corrective actions, the rod and bushing of the valve were replaced.

Report # 012500 PARTIAL FAILURE OF THE THREE PHASE SUPPLY SYSTEM.

One rotating DC/AC converter out of three for the three phase supply was out of service for maintenance. After the maintenance work, the workman tried to connect the converter and disconnected the synchronizing device for test. He was not aware of the fact that all relays in the feeding lines to the busbar have a common control voltage. This circuit test caused the loss of voltage on a busbar and resulted in the unavailability of the instrumentation in the control rooms. Two trains of the reactor protection system were also without power and by a 2 out of 3 logic, the reactor scrammed, MSIVs closed, auxiliary systems were isolated, diesel generators started and a trip of the cooling water, feedwater and condensation pumps followed. Within 3 minutes power was restored. The actions taken are: - Instructions to maintenance personnel to follow working procedures. - Improvement of the independent supply to the busbars.

Report # 012600 MALFUNCTION OF MASTER CONTROLLER IN FEED WATER CONTROL SYSTEM.

During the raising power operation at the last stage of the annual inspection and overhauling, turbine protection system was activated by the signal "high" of reactor water level. Then the turbine tripped and subsequently the reactor scrammed. It became clear that the incident occurred by malfunction of a master controller of feedwater system, and so the alleged master controller was replaced. There was no impact of radioactivity on the environment.

Report # 014800 FIRE RESULTING FROM TRANSFORMER FAILURE.

A fire occurred in the main-phase B transformer, which was carrying power from the plant, due to an electrical fault in the B transformer. The electrical fault ruptured the transformer, causing 9,000 gallons of oil to spray onto the transformer bay area. The spilled hot oil caught fire. Fire protection deluge systems for phase A and B transformers came on automatically and control room operators activated the phase C transformer system. However, the fire in the spare transformer area caused severe damage to the turbine building wall and the wall separating the phase B transformer from the spare transformer, the overhead bus bars going to the turbine wall, and cables running on the turbine wall. The fault in the B transformer resulted in a turbine trip because of the load loss. The reactor coolant pumps B and C and condensate pump A were secured to reduce station load. The main steam line isolation valves were closed to prevent further cooldown of the reactor coolant system. Numerous other plant disturbances occurred including diesels starting, safety injection and a decrease in air pressure. The fire which was caused by an internal fault plus inadequate maintenance and storage procedures was extinguished 1 hr. 20 mins. after it had started. [A related incident is described in IRS148.02]

Report # 016500 REACTOR SCRAM AND LOSS OF REDUNDANT SAFETY SIGNALS.

The BWR vessel water level is measured by a differential pressure measurement between a reference head provided by a condensate pot, and the head at a lower tapping close to the diffuser section of a jet pump. Due to the level transmitter equalizing valve being open, the reference head was lost and a high level signal generated. Several systems, some safety related, were affected by this spurious signal. It had been demonstrated that this level instrumentation was affected by the recirculation pump flow rate. It is stated that with the pump at full power there is virtually no pressure difference across the equalizing valve, and thus the cell, so the open valve does not affect signal. With flow at minimum there is a 2 psi pressure loss in the variable leg. The turbine tripped and reactor scrammed. Action(s) taken: Redesign/modification and control of similar equipment.

Report # 0176G4 INADVERTENT ACTUATION OF FIRE SUPPRESSION SYSTEM.

During startup testing of the new fire suppression system, an inadvertent actuation caused various power cabinets and electrical equipment in the turbine and intermediate buildings to be sprayed. A manual reactor trip was initiated following indication of two dropped rods and numerous control room annunciator alarms. The dropped rods were attributed to a trip of the "A" RPS MG set which may have reduced voltage enough to drop two rods. All systems functioned properly following the trip and the plant was maintained in "hot shutdown" status while operability of equipment affected by the suppression system was assured. Failure to follow test procedures caused actuation of several portions of the fire suppression system. All affected components were repaired.

Report # 019000 REACTOR SCRAM DUE TO MAIN STEAM ISOLATING VALVE CLOSURE.

The reactor was at full allowable power, when it scrammed due to the closure of the main steam isolation valves. Main steam line pressure was decreased to the MSIV closure setpoint due to malfunction of electrical pressure regulator servo valve. Following the reactor scram, the main turbine was manually tripped and condenser vacuum was broken for safety shutdown. Then approximately 20 Curies of radioactivity were released from the stack due to the increasing off gas flow temporarily. The reactor was planned to be restarted following the replacement and testing of the servo valve.

Report # 021600 LOSS OF REGULATION INCIDENT.

Reactor power is controlled by two digital computers, D1 for normal use and D2 for standby. Failure of a program in D1 should result in transfer of control of that program to D2 via the Data Link. The Executive Program had turned off the Reactor Regulating System Program, the Boiler Pressure Control Program, and an internal checking program (CHECK). The D1 program failures were then annunciated in the control room. However, the control was not transferred to D2 since the Executive Program on D1 continued to run. During the subsequent 36 seconds, the reactor power increased to the trip setpoint of 110% full power and the reactor tripped. Later, D1 was manually stalled to transfer control to D2. The fault was caused by a single bit, in the 16-bit D1 accumulator, failing to zero. Since the accumulator is used by the Executive Program to recognize and service interrupts from the various input/output devices, as well as from internal programs, failure of one bit resulted in inaccessibility to these devices/programs. A detailed investigation of the event is being carried out. Possible changes in hardware and/or software, to prevent a recurrence, will be identified. In the absence of protective system action, this event could have caused fuel failures and/or excessive overpressure in the primary heat transport system.

Report # 0218G2 STEAM EROSION IN TURBINE EXHAUST LINES

The unit experienced a 4-sq.ft. rupture of a 24-inch-diameter, long radius elbow in the feedwater heat extraction line which is supplied steam from the high-pressure turbine exhaust. The operator noticed the break in a steamline and tripped the reactor manually. The steam jet destroyed a non-safety-related electrical load center and non-safety-related instrumentation but did not render any essential equipment inoperable. Two persons suffered steam burns and were hospitalized. The rupture has been attributed to piping degradation that results from steam erosion. About 3 months prior to the failure, an ultrasonic inspection revealed substantial erosion of the elbow in the extraction line, however, the erosion was less than the licensee's criterion for rejection. The sustained reduced power operation and resultant lower quality steam were considered to contribute the acceleration of erosion and the failure of the elbow. The failed elbow and the identical elbow on the other feedwater heater supply line have been replaced. Similar erosion was found in the elbows and/or pipes of steam lines in Units 1 and 3.

Report # 0218G3 LEAKAGE FROM MOISTURE SEPARATOR DRAIN LINE

During routine inspection in the turbine building, a leakage was found from a moisture separator drain line. The leakage initially appeared as wetted insulation on the drain line, later, increased to the point where minor wisps of steam were blowing from the piping/insulation immediately under the moisture separator. The plant was shutdown manually to allow access, inspection and repair of the leak. Upon removal of the moisture separator drain line insulation, a crack and a through wall defect was found in a 6 inch diameter section of the drain line. Ultrasonic examination revealed extensive wall thinning. The pinhole defect was located at a point of minimum wall thickness. The extensive internal corrosion/erosion was found at the cut out section of affected pipe. The event is believed to have been caused by steam cutting associated with the two-phase fluid that flows from the moisture separator to the expansion header. The drain pipings for the other three moisture separators were examined and wall thinning was found. The affected piping (schedule 40) was cut out and replaced with schedule 80 carbon steel piping. Further, the licensee is considering including the drain piping in a surveillance program to monitor internal wall erosion.

Report # 0218G4 EROSION AND RUPTURE OF HEATER DRAIN PIPING, PARTLY DUE TO MISUSE OF PUMP DISCHARGE PIPE

The reactor tripped from full power following a turbine trip due to spurious high vibration signals on a main turbine bearing. The resulting main feedwater isolation produced, as expected, a pressure pulse in the heater drain and feedwater systems, which caused a rupture of heater drain pump discharge piping. Because of the steam and water buildup in the turbine building, condenser vacuum was lost approximately 4 minutes after the reactor trip. Loss of vacuum rendered the steam dump system inoperable, therefore, the power operated atmospheric relief valves (PORVs) were used to control steam pressure and plant temperature. The steam-water mixture of approximately 350 deg. F escaped into the turbine building, which actuated fire suppression (deluge) systems, damaged secondary plant equipment in the vicinity, and injured one operator. The rupture of the piping was caused by severe erosion/corrosion of the pipe wall. Because the feedwater flow out of a normally open 14-inch manual globe valve was directed against the pipe wall, the pipe wall at the rupture location had eroded from nominal thickness of 0.375 inch to that of 0.098 inch. Further, the pipe was not intended to carry full flow during normal full power operation, however, due to operational problems, the pipe did become the normal flow path. The damaged section of pipe was replaced. In addition, all secondary system piping runs subjected to temperatures in the range of 150-250 deg. C with flow rates greater than 15 ft/sec will be evaluated to identify flow perturbation areas where erosion/corrosion may be occurring.

Report # 0218G5 FEEDWATER LINE RUPTURE DUE TO EROSION IN AREA NOT REGULARLY INSPECTED

A pipe, immediately downstream of a feedwater heater level control valve, ruptured causing a steam leak. The reactor and turbine were manually tripped due to the possibility of grounding the steam generator feed pump motor(s) and/or heater drain pump motor(s) toward which the steam was blowing. The pipe rupture occurred because the flow, exiting the feedwater heater normal level control valve, impinged directly on the pipe surface and severely eroded the pipe in that area. The eroded section of the pipe was replaced. In addition, sections of pipe adjacent to flow control valve configurations, similar to the pipe that ruptured, will be included in the plant's reliability engineering program.

Report # 022100 BLACKOUT AFFECTING THE xxxx POWER STATION

The Unit 3 was at 87 percent of rated thermal power. Start-up tests were in progress. During a periodic test of the turbine protection system, a trip of the turbine occurred as a result of a spurious protection signal, leading to a reactor trip. A lack of reactive power production on the 380 kV grid due to the trip of the plant, caused a collapse of this grid. The normal fast switch-over of the load to the 150 kV grid failed causing an offsite power source loss for about one hour. All diesel generators started and loaded correctly. The computer had been automatically switched off 30 minutes after the reactor trip due to high temperature in the computer room caused by poor ventilation. The information relative to a 50-minute period was lost. A trip of Unit 3 caused grid instability then a trip signal to the main circuit breakers of Units 1 and 2. During the first seconds of island production oscillations in turbine speed caused a Reactor 1 trip on low frequency. As Unit 2 was at the end of its cycle, the first acceleration of the turbine led to a speeding up of the primary pump, an increase in the flow and a decrease of the average moderator temperature. This caused a rapid flux increase because of the very high negative temperature coefficient. The reactor trip on high flux followed within a second after the opening of the main circuit breaker. At the moment of the trip, the 150 KV grid was available but in a degraded situation. A low voltage signal induced the automatic start signal for the four diesel generators common to Units 1 and 2. One DG did not pick up the load and was backed up by another. The first back-up program using a third DG failed. Natural circulation and steam generator discharge to the atmosphere cooled the core for one hour.

Report # 022900 LEAKAGE IN THE CHEMICAL AND VOLUME CONTROL SYSTEM

While operating at full power, a routine changeover of charging pumps of the chemical and volume control system, CVCS, had begun by starting a second pump. The letdown flow temperature downstream of the regenerative heat exchangers became too high. The difference between the charging and the letdown flow showed a leakage of about 25 m3/h. A signal from the fire protection system in the primary containment led to automatic closing of the reactor building ventilation system, and the recirculation ventilation system. Water was detected in the sump, up to 20m3. A second charging pump was put in operation; the power was reduced, then the reactor was tripped. The cause of the leakage was a defective gasket in the flange connection of a safety valve in the charging line. These flat gaskets were replaced by new ones with a graphite clad cam profile in all relevant systems. This incident showed the need for air sampling systems in the generation area as well as in the installation area within the reactor building, in order to monitor the radioactivity in the course of an event. The total gaseous release was about 5.5 Ci of noble gases.

Report # 023100 FAILURE OF ALL THREE SEALS ON A PRIMARY PUMP

Commissioning tests were being done at about 75 percent power. An alarm signal showed high flow past the first of three shaft seals of a primary pump. This should have caused isolation of the return line from between seals 1 and 2, but because of a wiring fault the valve did not close. Four minutes later high flow past the second seal was alarmed. Again three minutes later an alarm showed water was passing the third seal. High temperature plus rise of sump level, then showed that primary coolant was leaking, past all three seals. The reactor was shut down and in five hours thirty cubic meters of water leaked out. Two types of previous incidents or test could explain the seal damage. Emergency plant cooling or emergency boration signals apparently lead to a pressure transient on the seals with possible harmful effects. During one emergency plant cooling test, some primary pumps ran for several minutes without supply to the seals or cooling of the thermal shield. Following the incident the three seals on each primary pump were inspected. A safety valve was fitted upstream of the isolation valve on the seal return line to limit pressure transient on the no. 2 seal. Modifications of the seal flow and return system are being investigated. Procedures will be improved.

Report # 023600 FAILURE OF ELEVEN SAFETY RELIEF VALVES TO ACTUATE AT SETPOINT.

A spurious reactor high pressure signal resulted in a reactor scram. The main steam isolation valves

(MSIVs) closed automatically when reactor water level decreased -30 inches. The high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) started automatically. Reactor water quickly recovered and upon reaching the high level setpoints, the HPCI and RCIC tripped off. Reactor pressure began increasing with the MSIVs closed. The 11 safety relief valves (SRVs) which were supposed to lift between 1080 and 1100 psig did not actuate to terminate the pressurization. 12 minutes after the scram, three of the SRVs lifted. The MSIVs were manually opened, the electrohydraulic pressure control system, which admits steam to the turbine to control pressure, took over and normal recovery was initiated by the operators. The most likely cause of the failure is some combination of friction in the labyrinth seal area and/or sticking valve of the pilot disk in its seat. Later, the disassembly of the sticking valve revealed significant corrosion or sediment in the disk/seat area. The pilot sections from all 11 SRVs were replaced. The old pilot sections were sent for laboratory test. 9 out of 11 valves will be exercised regularly. 2 valves will not be exercised and will be utilized for possible future testing. The valves in Unit-2 will be exercised in the same way.

Report # 0241G1 FROZEN INSTRUMENTATION LINES CAUSE INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES

Engineered Safety Features (ESF) were actuated inadvertently, causing safety injection, steamline isolation, and reactor and turbine trips. ESF was automatically actuated as a result of a two-out-of-three low pressure coincidence trip of steam generator (SG) "A" pressure instrumentation channels. Both channels tripped when pressure was released from their impulse lines through manually cracked open test connections. This was attributed to the frozen instrumentation lines due to the cold weather. Two of 4 SGs' instrumentation lines were affected by the cold weather since the housing for these lines was a concrete structure with an uncontrolled atmosphere (exterior). As an immediate corrective action, tracing heat was applied to the instrumentation lines in the exterior concrete housings. Further, the following permanent solutions were considered; permanent trace heating; a comprehensive cold weather procedure; or providing adjustable dampers on the exterior housings.

Report # 024702 PRIMARY PUMP SEAL FAILURE

The unit was starting up after refueling. During the refueling shutdown, primary pump seals had been inspected, and one seal n.1 changed. The flow past the replaced seal was high but acceptable. Later a high leakage flow rate alarm was received from the same seal, and then a high level alarm from the equilibrium tube (standpipe), at the outlet of the seal n.2. Leakage increased causing low pressurizer level, rise of containment pressure and high level in the normal containment sump. The plant was brought to cold shutdown. Total loss of primary water to containment sump was 80 cubic meters. Some gaseous releases occurred during the transfer of contaminated water from the reactor building sump to the effluent collecting tank in the auxiliary building. Several destructions were found on each of the three seals of the pump. An incorrect remounting of the "O" ring between the aluminum oxide ring and its support could be at the origin of the incident. During the incident two alarms appeared from the fire detection system. In addition to the repairs, studies are underway relative to: - the action to be taken in case of increase of the seal no. 1 leak rate - the opportunity of inhibition of the automatic isolation of the thermal barrier coolant system.

Report # 025800 LOSS OF AUXILIARY POWER AFFECTS TWO UNITS.

Units-1 and-2 have three diesel generators (DGs), DG-1 for Unit-1, DG-2 for Unit-2 and DG-1/2 for either Unit-1 or-2. DG-1 was out of service for maintenance. While preparing to remove the Unit-2 reserve auxiliary transformer from service for repairs, an equipment operator mistakenly pulled the fuses for a 4-KV bus, followed by a reactor trip and loss of all normal AC power to Unit-2, then both DG-2 and DG-1/2 started automatically. 22 minutes later, when the residual heat removal (RHR) service water pump was being started, DG-1/2 tripped including numerous alarms and loss of control room indications. Further, the loss of DG-1/2 left Unit-1, which was still operating, without any onsite emergency AC power. As the pressure in the Unit-2 containment increased from 1.3 psig (normal) to 4.3 psig, emergency core cooling was initiated. Thirty-nine minutes after the event began, offsite power was restored. The cause of total unavailability of emergency DGs for Unit-1 could be attributed to nonconservative planning of maintenance activities, operator errors, and design error. The trip of DG-1/2 was caused by a design error in the DG control logic system. The under excitation relay that caused the DG trip has been removed, and the licensee is planning modifications to all diesel generators to prevent protective trips in an emergency situation. The leaking gaskets on the relief valve discharge line, which had contributed to the rise in containment pressure, were replaced.

Report # 026200 MALFUNCTION OF MAXIMUM FLOW LIMITER IN FEED WATER CONTROL CIRCUIT

The power station was operating at full power. Due to a malfunction of maximum flow limiter in the feedwater control circuit the reactor shutdown automatically by the signal of low water level. The limiter was replaced and the plant returned to service.

Report # 026300 REACTOR TRIP DUE TO INADVERTENT OPERATION ON POWER SUPPLY TO THE CONTROL ROD DRIVE MECHANISM

The reactor was automatically tripped from full power operation due to actuation of a power range neutron flux negative rate trip signal. Investigation showed that the power source for the control circuit of control rod drive mechanism was inadvertently tripped during maintenance work of the troubled power unit, causing the insertion of some control rods. The troubled power unit was replaced and the control circuit was checked before resuming power operation.

Report # 027002 DRAIN OF PRIMARY WATER DUE TO MISALIGNMENT OF A RHR VALVE

During the refueling of the plant, while the overall leakage test on the primary containment was in progress, a maintenance operator inserted, mistakingly, the power supply plug of the valve located at the suction from the suppression pool of the division 1 of the RHR system, in shutdown cooling mode. This fact caused the immediate opening of the valve due to the incorrect position of the related switch in the control room. 20 cubic meters of primary water drained from the reactor pressure vessel to the suppression pool. The reactor scrammed on low level and automatic isolation of the RHR system occurred simultaneously, concluding the event. The technical procedure for the containment leakage test required the RHR to be lined up with the suction valve closed, but because of the disconnection of the electrical power for previous maintenance, the closure of the valve had been actuated by hand and the switch had been left on the "open" position. The following human errors were identified: - insufficient analysis of the compatibility of work orders with the particular situation of the plant. - incorrect positioning of the switch in the control room. It shows the following quality-assurance deficiencies: - defective application of the procedure - defective application verification Considering these errors as isolated errors, no specific corrective actions have been prepared. This is a follow-up report to IRS 270.00

Report # 029000 EMERGENCY BUS LOSS DUE TO BREAKER PROBLEMS

After finding a steam and water leak of 20 to 40 gallons per minute in a cracked weld of a non-safety related heater drain pipe, an orderly shutdown of the reactor started from 60% power. With the reactor at 17% power, both manual and automatic transfer of electrical feed to a bus failed. Further, a diesel generator for the emergency bus also failed to close on the emergency bus. This caused a loss of voltage to an emergency bus followed by a Group 1 primary containment isolation and a resultant reactor scram. It was found that the unit startup transformer (SUT) output breaker indicated a problem in closing the breaker. The interchangeable breaker of the other transformer was installed, and power to the bus was restored. The failure of the breaker was caused by a sheared breaker charging spring motor actuator. This prevented charging of the breaker charging springs for breaker closing capability. While the unit was in cold shutdown pending repair of the cracked heater drain pipe, the investigation of the output breaker of the diesel generator was conducted and revealed that simultaneous close and open signals to the breaker prevented automatic closing. As counter measures, operational and surveillance procedures, and the design were modified.

Report # 029600 REACTOR SCRAM DUE TO EXCESSIVE VALVE LEAKAGE

The reactor was manually scrammed on December 15, 1982 when significant abnormal piping vibration was observed on one of three feedwater systems during a plant start up. When the feedwater pressure increased, leakage of cold feedwater back through the string isolation valve and the feed regulating valve mixed with the hot steam/water mixture causing significant water hammer. Plant startup procedures were revised to ensure that the idle feedwater strings are maintained pressurized. The leaking valves were inspected and repaired. - The excessive valve leakage in the feedwater system contributed to a reactor scram on December 15, 1982 from about 6% power on low water level while the reactor water cleanup (RWCU) system was being put in service. When the RWCU system inlet valve was opened, the feed string outlet valve was shut. The sudden increase in outflow from the reactor without an available feed path caused a drop in reactor water level and a reactor scram. The licensee planned an extensive valve overhaul program.

Report # 030302 TROUBLE WITH ELECTRICAL SUPPLY SYSTEM CAUSED BY LIGHTNING

While operating at full power, there was a lightning stroke around the main stack. The standby power supply breaker trip alarm was generated and the power generation went down to zero. As there was no alarm for the reactor, the reactor scram was confirmed on the neutron monitoring system. Inspection of the main control room and major operation panels revealed a melted fuse of the power supply for the reactor water level meter that led a signal requiring increase of feedwater and resulting in turbine trip and the following reactor scram. Also some electronic parts with low insulation-proof and connected to the DC 125 V or the AC 120 V power supply for instrumentation were damaged. The lightning surge entered the in-building grounding system via the exhaust pipe or the exhaust duct to raise the potential of the grounding system. In addition, the lightning surge flow through the ground and grounding wire increased by induction the potential of the neighboring control instrument cable. The following measures were taken to prevent recurrence: - improved grounding of the main stack, - improved grounding of the main control panel, - improvement of measures for the outdoor control instrumentation system.

Report # 031900 IMPACT OF A LIGHTNING STROKE INTO PLANT'S 220 KV LINE

Lightning struck the main external power line between the plant and the switchyard. Protective devices opened the plant's switchyard breaker. However, contrary to the plant's lightning protection design, electronic devices in the turbine generator control system were destroyed. The lightning induced an overvoltage in a signal line between the switchyard and GKN. The overvoltage destroyed the fast fuses located at the end of the line, the transmitter module processing the signals, three other modules in the same transmitter cabinet, the Zener diodes for protection against overvoltage, another four modules in an electric cabinet. As a result of the lightning stroke, a spurious signal was generated which caused a faulty switch over of the turbine generator voltage controller from automatic to manual operation. This had the effect that, after the automatically actuated load reduction to onsite power, the generator excitation was not settled back and that the generator protection device was triggered by the signal "high generator voltage." The generator breaker was opened by this device, thus causing a loss of offsite power, since by current design an automatic switch over to the reserve grid was not initiated by a criterion derived from a high generator voltage. The following remedial actions have been performed during the last refueling outage: - complete underground bedding of outside cables - use of varistors and isolating transformers - in respect to the switch over to the reserve grid an additional criterion is used. If the voltage of the service station bus bars decreases to a certain value, a fast transfer to the reserve grid is activated.

Report # 032100 SHORT CIRCUIT IN THE STATION SERVICE SUPPLY CAUSED BY HUMAN ERROR AND SUBSEQUENT FAILURE OF A STATION SERVICE TRANSFORMER

For a revision of an electrically heated auxiliary boiler, the power supply of the boiler and the boiler itself, for safety reasons, were grounded. After the revision work, the ground wire of the boiler itself was not removed. Therefore, when the boiler was reswitched to the station service bus bar, a short circuit was formed. The short circuit caused, as designed, the opening of the supply breaker, but led to an unforeseen failure of the station service transformer activating the transformer protection system, followed by the opening of the generator breaker and the 220 kV breaker, and a reactor trip. To reduce this type of human error, analysis and optimization were performed to aid coordination of the individual measures taken in the course of putting components out of and back in service. The failure of the station service transformer was caused by a high current during the short circuit, which induced high radial and axial forces on the windings of the transformer resulting in the deformation and a short circuit in the windings. Further, it was found that the supports for the windings were not installed in the failed transformers. This was attributed to a mistake in the construction drawing. The damaged windings were replaced and the supports were installed.

Report # 032700 MAIN FEEDWATER LINE BREAK DUE TO WATER HAMMER.

At the plant, upgrading of main and auxiliary feedwater systems had been performed recently. During full power operation with one turbine-driven feedwater pump (FWP), (two motor-driven FWPs were out of service), a reactor trip occurred, followed by turbine and turbine-driven FWP trips. This resulted in a complete loss of normal feedwater flow and a decrease in boiler level. Auxiliary feedwater system was actuated automatically. Approximately 15 minutes after the trip, severe water leakage occurred in the pipe adjacent to the weld joining the pipe and SG No.2 safe end. The leakage continued for 12 hours at a maximum rate of 100 gpm. The cause of the leakage was water hammer which caused an existing crack to propagate through the pipe wall. The location of the crack coincided with a stress area in the pipe where previous PWR experience had identified the likelihood of thermal stress cracking. Similar cracking had also begun on other SG's pipes. The cause of the water hammer is attributed to incomplete consideration in ongoing design and operational plant upgrading. Cold auxiliary feed water can cause high thermal stresses and water hammer due to rapid steam condensation. Damaged piping and supports were repaired. A design change was implemented adding J-tubes to the top of the SG feed rings. A number of operational changes also were made.

Report # 033000 INADVERTENT RPS TRIP WITH PORV ACTUATION.

With the unit in hot shutdown, testing was in progress on an inverter in the 120V vital AC system, which supplies power to the reactor protection system (RPS). In order to avoid blowing fuses when the inverter was returned to service, instruction was given to de-energize the related channel of the RPS. But the wrong channel was de-energized in error. (The RPS has 4 channels) When the inverter was reconnected, the fuses blew and the channel was de-energized. Due to loss of 2 out of 4 channels the RPS was actuated and the power operated relief valve (PORV) opened. The resulting rapid pressure drop caused an engineered safety features actuation signal. But no water was injected. Although the rupture disk in the quench tank opened no significant quantities of water on radiation were rebased. The fuses blew due to crossed power leads in the inverter blew due to crossed power leads in the inverter. The leads were wrongly located during maintenance. Some procedures were changed to prevent a recurrence.

Report # 035700 LEAKAGE FROM THE SEATS OF PRESSURIZER RELIEF VALVES

The reactor, being operated at full power, was automatically tripped due to "high neutron flux rate" which was caused by a transmission line trip from lightning. After the trip, an increase in temperature at the outlet of the pressurizer relief valve was observed and it suggested some leakage from the seats of those valves. The investigation revealed that rubbed scratches were found on each contact surface of the two plugs and cages of the valves. Therefore cages and plugs of the two valves were exchanged for new ones. After having certified the integrity of the two valves, the plant was returned to service.

Report # 035800 REACTOR SCRAM DUE TO LOSS OF DC POWER SUPPLY.

The power station scrammed when it was operating at 140 MWe. It was revealed that the scram was caused by the loss of DC power supply to retain the control rods, due to the burnout of one contactor coil in the circuit. After replacing the coil for a new one and confirming the circuit integrity the plant was restarted.

Report # 037500 UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT

During power operation, an unexpected closure of one main steam isolation valve (MSIV) led to reactor trip and the safety relief valves (SRVs) open. Reactor core isolation cooling (RCIC) system was used for level control. After the reactor was stabilized, the scram was reset, this caused the scram discharge volumes

(SDVs) to begin draining. During the reactor pressure was maintained with SRVs, one of the SRV tail pipe vacuum breakers failed and allowed steam release into the drywell, resulting in a second scram. In spite of scram signal, the SDV drain line isolation valve did not fully close, providing an open flow path between the reactor coolant system and reactor building equipment drainage (RBED) system. A missing hub cover on RBED system allowed hot steam to escape into RCIC room. The resulting high temperature caused a spurious instrument signal and the RCIC system tripped. The reactor water level was maintained by restarting feed water pump. After the drywell pressure was reduced with the drywell chiller (which was started by bypassing trip signal), the reactor scram was reset and the coolant leakage was terminated. - The MSIV failure was due to the main valve disk separation from the valve stem. The MSIV lock pin will be installed. - The SDV's isolation valve failure was due to loose valve body-to-operator yoke. The scram discharge system will be modified. - The hub cover on RBED system had been removed improperly during maintenance work. Administrative controls will be upgraded. - The SRV failure was due to stuck open of tail pipe vacuum breaker. The breaker design will be changed

Report # 039000 REACTOR DEPRESSURIZATION RESULTING FROM FAULTY STEAM RELIEF VALVE.

The reactor was being shut down for planned maintenance and surveillance activity. After the generator was taken off line and with the reactor at about 20% rated thermal power a routine surveillance test of the three-stage Target Rock steam relief valves (SRVs) was performed. The valve opened but did not reseat fully. This reseating failure lasted 14 minutes. The changes of the various parameters during the incident were as follows: - Reactor pressure decreased from 960 psig to 360 psig. - Coolant temperature decreased 85 deg. F in 20 minutes. (Technical Specification limits: - 100 deg. F/60 min). - Suppression pool temperature reached 101 deg. F (T.S. limits: 110 deg. F). The reactor was manually scrammed during the incident. The cause has been traced to second stage pilot-seat erosion due to steam cutting and/or particle hang-up at the seat-disc interface. The safety valve was replaced. Similar spurious blowdown of the Target Rock SRVs has been reported. Many BWR plants have replaced the SRVs with newly designed two-stage SRVs. This problem has identified with only the Target Rock's SRVs.

Report # 039600 NON-CLOSURE OF SAFETY RELIEF VALVE.

Erroneous generator under-excitation caused a disconnection of the plant from the 380 kV grid. Since the short term switchover of the station service buses could not be performed due to voltage mismatch, the long term switchover was induced. Simultaneously, reactor scram and turbine trip occurred. The condenser circulating water (CCW) pump, which should have remained in operation, tripped due to low bearing oil pressure. This caused the closure of turbine bypass station and thus a total loss of heat sink. Safety valves opened due to high reactor pressure but one of them failed to reclose and reactor pressure dropped rapidly. After this, reactor behaviour was as designed. The relief valve problem was attributed to the loosened prolonging cap of the lift magnet stem in a pilot valve. In addition, the pilot valve had the original problem that the valve spindle could not reach its end position. Vibration during transportation seemed to contribute. The trip of the CCW pump was caused by insufficient delay time of "oil pressure low" signal rejection. Corrective actions: - Pilot valves: Transport protection and inspection will be improved. Longer tightening screw will be used in order to limit the spindle lift. - CCW pump: The reserve oil pump will be DC powered.

Report # 040600 REACTOR SCRAM DUE TO MAIN STEAM ISOLATION VALVES CLOSURE

The power station was operating at full allowable power. Main steam pressure controller was being switched from mechanical to electrical pressure regulator. The electrical regulator was affected by noise from the first stage amplifier and gave false response causing rapid opening of electrical pressure regulator servo valve and turbine control valve. Low steam pressure due to rapid opening of turbine control valves caused closure of MSIVs and automatic reactor scram. As a temporary repair, the amplifier was replaced with a spare one. Thereafter, as a permanent countermeasure, during the periodical inspection outage, the electrical pressure regulator unit was replaced with a new one that had noise protections.

Report # 042503 VOLTAGE DROP FOLLOWED BY LOSS OF TRAIN A 48 VOLT BUS LEADING TO LOSS OF BOTH OFF-SITE POWER AND TRAIN A DIESEL GENERATOR

During normal operation, shortly before midnight, a control card failure caused the loss of train A 48 V, DC rectifier. The buffer battery held the 48 V supply for several hours but, finally, ran down to 30 V. The low voltage alarmed in the Control Room but was not noticed since it was grouped with five other less important ones related to faulty insulation defects) which had been flashing frequently. The low voltage caused: - Train A of reactor trip breakers to open - Failure to transfer to the auxiliary off site supply - Failure to start the train A emergency diesel generator. Train B diesel generator started and fed the essential systems. Electric power transients led to a 220 V safety fuse to blow, which led to the loss of some information in the control room and to primary pressure transients. The primary pumps stopped, the setting valves of the CVCS feedline and for the seal water injection line fully opened, while the setting valves of the CVCS relief line closed. The pressure in the primary system was then controlled by the B train pressurizer discharge valve which was actuated several times. The situation was brought under control after one hour by manual setting of the circuit breakers and power supplies, as no procedure allowed to cope with such a situation. There was no external release and no effect on fuel.

Report # 0437G3 TEMPORARY LOSS OF OFFSITE AND ONSITE AC ELECTRICAL POWER.

The plant experienced an event involving a temporary loss of all AC power including failure of the emergency safety system buses. With the unit operating at 30% power, the licensee was conducting planned startup testing at the time of the incident. This report was followed by IRS0437.G5 (original numbering is 437.2 (III) distributed in 1985, May, 31st). Concerning detailed description, see IRS0437.G5.

Report # 0437G5 TESTING RESULTS IN STATION BLACKOUT

The plant operated at 30% power and preparations were being made to carry out the startup test "Loss of Turbine Generator and Offsite Power". During the isolation of the Unit-2 electrical distribution system from the Unit-1 system, wrong DC supply knife switches were opened, which shut off DC power to all the emergency safety system (ESS) logic circuitry. (This resulted in loss of many functions including (1) automatic transfer of ESS buses to alternate power source; (2) automatic diesel generator (DG) startup; (3) ability to re-energize 4KV ESS buses from an offsite source; (4) automatic load shedding; (5) low pressure ECCSs). This mistake was not noticed in spite of the operation being supervised and the presence of indication on cubicles. The startup test was initiated and "Loss of Turbine Generator and Offsite Power" condition was established. Consequently total loss of AC power occurred. In addition to the above mentioned function loss, most instrumentation in the control room failed down scale. Abortive attempts were made to manually startup the DGs and energize the buses. Eventually the 4KV buses were re-energized from the Unit-1 transformer. The cause is attributed to operator error, inadequate training, imprecise procedures, ineffective verification and inadequate implementation of corrective actions for previously identified problems. [This report is a follow-up of IRS0437.G3].

Report # 044300 WATER HAMMER IN FEEDWATER PIPING AND SUBSEQUENT SCRAM DUE TO FEEDWATER SYSTEM PROBLEMS

The reactor was in hot standby and stroke tests were being performed on the four feedwater regulating valves. A main feedwater line check isolation valve had apparently failed to open. When its associated regulating valve was stroked open the check valve slammed shut which generated pressure waves resulting in water hammer and causing the following damage: - Major distortion of the piping support system - Distortion of a cam positioner of a control valve and a position transmitter of a by-pass valve - Distortion of trunnions on the feedwater piping in containment - Plugging of a flow transmitter line with rust buildup. Review of the event revealed procedure for valve lineup had not been completed prior to the tests. Two attempts to start up were hampered by turbine and reactor scrams on high boiler level. Subsequent investigations revealed problems with feed flow indication and the dislocation of a steam generator feedwater nozzle, apparently caused by water hammer. The latter was not discovered in the post-scram review. Actions taken include testing and visual inspection of feedwater lines, modeling of the incident, repair and replacement of parts, and improvements to the quality of post-scram review. The lesson learned is that after water hammer incidents inspection must cover all equipments.

Report # 044400 REACTOR SHUTDOWN DUE TO INOPERABLE HPCI SYSTEM AND SAFETY RELIEF VALVE FAILURE.

The plant was operating at 88% power. After successful high pressure coolant injection (HPCI) pump operability surveillance, the pump lubrication oil was found to be contaminated with water. The HPCI system was declared inoperable and testing of redundant safety systems was immediately initiated. During the automatic depressurization system (ADS) surveillance, an electromatic relief valve failed to open. The reactor was shutdown in accordance with technical specification. The cause of HPCI pump oil contamination was a leak in the pump's oil cooler system. The failure of the relief valve was caused by a disconnected coil in the valve controller. High vibration levels of the valve caused the disconnection. The HPCI pump oil cooler O-ring and gaskets were replaced with new parts. Concerning the relief valve, a modification to ensure a more positive connection was performed.

Report # 049500 HIGH PRESSURE COOLANT INJECTION SYSTEM LOCKOUT.

While one of the MSIVs was undergoing a partial closure test, it continued to shut instead of returning to open position. In spite of immediate power reduction, this caused a reactor scram and operation of primary relief valve. The water level in reactor vessel reached a high of 184 inches during the transient. The pneumatic pilot valve assembly, that caused the MSIV problem, was replaced. On April 20, 1984, while the plant was operating 100% power, the monthly high pressure coolant injection (HPCI) valve operability test was performed twice unsuccessfully. The Senior Control Room Operator correctly assumed that the high vessel water level trip signal in the HPCI logic circuit (due to high water level condition during the transient on April 16) was still sealed in, locking out the HPCI system. The trip signal was cleared and the HPCI system testing was performed successfully. No direct indication/alarm of a HPCI lockout condition is provided in the control room. Plant start up procedure does not provide individual sign offs for each reset button. The related procedure has been modified. This experience will be taken into consideration by the plant control room design review committee.

Report # 051604 LOSS OF MAIN AND AUXILIARY FEEDWATER SYSTEMS

During 90% power operation with the No. 2 main feedwater pump (MFP) operating in manual control, No. 1 MFP tripped on overspeed due to its automatic control problem. Thereafter the reactor and turbine tripped automatically. Soon after the reactor trip, the steam and feedwater rupture control system (SFRCS) activated due to a spurious signal, causing the main steam isolation valves (MSIVs) to close. As a result, No. 2 MFP coasted down (i.e. complete loss of main feedwater), causing the decrease of once-through steam generator (OTSG) inventory . Although the SFRCS was actuated due to OTSG low level, an operator tripped SFRCS manually and inadvertently set it improperly. This caused the isolation valves in each auxiliary feedwater (AFW) train to close. Due to improper settings of the torque switches, these isolation valves did not reopen automatically even after the SFRCS reset. In addition, both AFW pumps which had already started tripped on overspeed. Pressure and temperature in the reactor coolant system continued to rise and the pressurizer pilot operated relief valve (PORV) opened three times. On the third lift, the valve remained open. The operator closed the PORV block-valve and reopened it. Fortunately the PORV closed. 16-18 minutes after the reactor trip, the startup and auxiliary feedwater pumps started and the isolation valves opened. Thereafter, the plant shifted to a stable condition. During the incident, make-up/high pressure injection cooling was not initiated in spite of the emergency procedures. As described above, the cause of the incident was a number of equipment malfunctions and operator errors. In addition, licensee's insufficient care of the plant equipment is pointed out.

Report # 052000 LOSS OF ELECTRICAL POWER TO A UNIT xxxx

A fault on a 13.8 kV power supply bus tripped the differential protection (87) on the Unit 4 system service transformer. The unit suffered a complete loss of Class IV power (power which can be interrupted without safety systems being affected) as the bus fault was cleared and isolated. The loss of power to the heat transport pumps caused the unit to trip on heat transport low flow. The initiating cause was due to contact of a flexible metal conduit, which carried current transformer secondary wiring, with the busbar epoxy insulation of one phase within a disconnect switch cubicle. This resulted in fretting of the insulation, and ultimately a short to ground. The current transformer chamber will be inspected on all units to determine if fretting of bus insulation by current transformer secondary wiring is a generic problem.

Report # 052300 EXPLOSION AND FIRE IN AUXILIARY TRANSFORMER RESULTS IN LOSS OF STARTUP TRANSFORMER

During 56% power operation the auxiliary transformer (AT) failed, causing turbine trip and reactor scram. A resulting fire ball caused damage and carbon deposits on two high voltage lines. In addition, the subsequent carbon deposits and insulator damage on the startup transformer (ST) caused the ST to trip on a phase-to-ground fault. After the ST trip, the system almost responded as designed and the reactor was stabilised. During this incident the emergency notification system hot-line was inoperable due to the loss of non-essential power. Recovery operations proceeded as expected, although some radiation monitorings alarmed spuriously and fire resulting from the AT failure occurred at the transformer and at the turbine building. Concerning the fire, the fire wall between AT and STs limited damage to the ST. The cause of the AT failure was a short circuit between turns in the high voltage winding. Resulting arcing generated gas at a rapid rate and pressure built-up in the transformer tank too quickly for pressure relief, which caused the tank to rupture. The failure experienced was not preventable by presently known or used maintenance and inspection technique. Corrective actions, including the following, were taken: - transformer testing was reviewed in order to ensure use of the best applicable oil and gas-in-oil analysis. - the reliability of the various telephone systems is being evaluated.

Report # 052600 REACTOR TRIP DUE TO AIR LINE FAILURES CAUSED BY FATIGUE CRACKING

On November 5 and 6, 1984, at the plant, air lines supplying the steam generator (SG) feedwater regulatory valve failed, resulting in actuations of engineered safety features and reactor trip. - On November 5, the reactor tripped from 45% power due to low-low signal on SG-C. The reactor trip was accompanied by a turbine trip, feedwater isolation signal (FWIS), auxiliary feedwater actuation signal (AFAS), and SG blowdown isolation signal (SGBIS). The cause was that the service air line manifold common to the SG feedwater control valves (FCVs) sheared. - On November 6th, a high signal on SG-C resulted in the turbine trip, FWIS, AFAS and SGBIS. A low-low signal on SG-D caused a reactor trip. The cause of the incident was the rupture of the service air line to the feedwater recirculation valve. The failure triggered feedwater oscillation and subsequent high level signal on SG-C. Both air line failures were due to improper material application, which resulted in fatigue cracking caused by the vibration imposed on the air lines during feedwater system operation.

Report # 053200 ANTICIPATORY REACTOR TRIP ON GENERATOR FIELD BREAKER TRIP CAUSED BY WIRE IN AMPHENOL CONNECTOR

During 43% power operation, the generator lockout relays and generator field breaker opened, causing a turbine trip and subsequently an anticipatory reactor trip. During the trip recovery, two main steam relief valves (MSRVs) did not reset properly. As a result, main steam pressure had to be dropped to approximately 880psa to reseat the valves. Investigation revealed that a wire in the electrohydraulic control system was loose at the amphenom connector in the electrohydraulic junction box. This loose wire triggered the turbine trip. The cause of the MSRVs delayed reseating is not fully known at this time, but MSRV lift set-point adjustment procedures were improved.

Report # 053900 SWITCHYARD COMPUTER DESIGN DEFICIENCY CAUSES LOSS OF NORMAL OFFSITE POWER.

The reactor tripped due to an electrical disturbance when two transmission line power circuit breakers (PCBs) tripped on overcurrent. The switchyard computer was being returned to service after maintenance. When the computer control outputs were re-enabled the switchyard PCBs opened. (The control output relays should have been open before the control outputs were enabled.) As a result, all the unit generator outputs were concentrated on the two lines, resulting in overcurrent. The cause of the incident was a component malfunction/failure, due to which the control circuits were changed to an undesirable state without a command from the computer. A design deficiency also contributed because the computer program did not include the function to reset the computer output control circuits when the computer is restarted. Corrective actions, including the following, were taken: - Addition of a control output relay test circuit in order to confirm that all control output relays are open prior to re-enabling. - Modification of the computer software in order to reset all control output relays on re-initialisation. - Increasing relay settings on the two tripped lines in order to allow these lines to carry full unit output.

Report # 055000 REACTOR TRANSIENT CAUSED BY GENERATOR BREAKER FAILURE

During 65% power operation, the turbine tripped. Subsequent switchover of auxiliary power supply caused the 10KV-grid (to which main feedwater pumps are connected) feed breaker to trip and the dump blocking valves to close leading to no by-pass to the condenser, resulting in a reactor trip. While the main feedwater pumps (MFWPs) were restored, the voltage of the 10KV grid decreased momentarily due to high starting current. This voltage drop caused the control valves of the pressure relief valves to trip. As a result, two pressure relief valves opened partially, which was not recognised for a while. Thereafter, some transient of reactor water level and pressure was induced. The cause of the turbine trip was overheating of one phase in the generator breaker. Overheating was caused by a mechanical part with faulty material. The cause of the dump blocking valve closure was attributed to insufficient nitrogen gas pressure in the hydraulic oil pressure system for controlling the valves. The pressure relief valve failures might not have been evident in the beginning because of inadequate fault indication. In addition to countermeasures for the above points, the automatic control of the valves has been modified so that the motors start at returning voltage.

Report # 0559G2 REACTOR SCRAM ON LOW CONDENSER VACUUM WITH SUBSEQUENT MSIV FAILURES

The operators were swapping the steam jet air ejectors (SJAE) when the reactor scrammed due to a turbine trip on loss of of condenser vacuum. That moment the plant was at 49% power. SJAE A was being placed in service due to apparent intercondenser fouling of SJAE B, unable to maintain condenser vacuum. The breaker for suction valve F003B on the B train was tagged open for maintenance work. However, with the A train supply valve (F505A) fully open, the steam supply pressure to SJAE A was 60 psig. In order to increase the steam supply pressure, the B SJAE supply valve F505B was throttled closed. The steam supply pressure increased to 115 psig. At this time, the operators noticed that condenser vacuum was decreasing. The A train suction valve F003A, which needed to be open to increase the vacuum, had dual indication. The valve was given an open signal both locally and remotely, but did not respond. An operator was sent to open valve F003A while another was sent to clear the tag and close SJAE B suction valve F003B. During this time, condenser vacuum decreased to the turbine trip setpoint and a turbine trip occurred followed by a reactor scram. The failure of the MSIVs to stay closed after being shut with the test circuit, and their inability to be closed normally, indicated a failure of the dual solenoid valve in the automatic actuation circuit. Corrective actions included (1) a design change to increase the size of the SJAE steam supply piping and to complete other SJAE enhancements, (2) cleaning the B intercondenser, (3) implementing an NRC-approved action schedule for increased exercising of the MSIVs prior to start-up, and (4) replacing the MSIV dual solenoids. Other necessary corrective actions will be completed after the root cause is determined.

Report # 056300 INOPERABLE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) FOLLOWING LOW

VESSEL WATER LEVEL CONDITION

During 61% power operation, the unit scrammed due to a reactor low water level trip signal as a result of a vital AC power supply trip, which caused both reactor feedpumps to run back. The reactor water level decreased approximately 100 inches and Groups 2 and 5 isolations were received. High pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) automatically started to recover the reactor water level. One of the tripped feedpumps was started manually, therefore the HPCI and RCIC were secured. The high water level turbine trip was then received for the HPCI, RCIC and reactor feedpump turbines. Reactor pressure remained steady with electro-hydraulic control (EHC) system. Thereafter, the reactor water cleanup (RWCU) outboard isolation valve was opened to re-establish RWCU system flow in order to accurately determine the vessel bottom head temperature. It was found out that the temperature difference between the steam dome and the bottom head was too great and the recirculation pumps could not be started. It was decided to depressurise the vessel. While the reactor pressure was being lowered, the high reactor water level tripped the feed pumps, resulting in reactor water level decrease and actuation of the reactor protection system logic. During this incident, both RCIC and HPCI were declared inoperable. It is believed that the actuation of undervoltage relay ETRIA was the most likely cause of the reactor scram because several other loads (in addition to vital AC) were lost simultaneously from the 600V bus. The cause of RCIC inoperability was trip/throttle valve failure due to the looseness of centrifugal trip weight's spring. The cause of HPCI problem was the turbine stop valve being in the midposition due to a galled stem.

Report # 056600 REACTOR SCRAM ON LOSS OF CONDENSER VACUUM DUE TO FAILED EXPANSION JOINT, AND SUBSEQUENT HIGH CONTAINMENT PRESSURE

The reactor scrammed due to low condenser vacuum resulting from a failure of a rubber expansion joint connecting the turbine casing to the condenser. Closure of main steam isolation valves caused reactor pressure to increase. Reactor core isolation cooling and thereafter additionally high pressure coolant injection were started in order to control pressure. Subsequently, high primary containment temperature and pressure (HPCTT) condition occurred and related safety features were actuated. The cause of the expansion joint failure is brittleness due to ageing. The cause of the HPCTT condition was insufficient capacity of the reactor building closed cooling water (RBCCW) system. Only one heat exchanger was in service because of the extreme cold river water temperature and concern for recirculation pump seal embrittlement. The reactor water discharged to condenser during the incident caused additional heat load on the RBCCW system. It is under review whether to increase the high drywell pressure trip setpoint in order to reduce the spurious trips.

Report # 057000 INOPERABLE SAFETY INJECTION PUMPS

The reactor was in hot shutdown and the emergency core cooling system accumulator tanks were being filled by a safety injection (SI) pump. during this task all three SI pumps were found to be inoperable. Investigation revealed the following points: - The cause of the inoperability of the three SI pumps is attributed to boric acid precipitating from a highly concentrated solution, solidifying and preventing suction flow. The source of the boric acid is believed to be the Boron Injection Tank (BIT). In this plant, the BIT discharge line is aligned to the suction of the SI pumps. - An additional cause of the inoperable SI pumps was gas, which caused the pumps to bind. The major constituent of the gas was nitrogen, but the source has not yet been identified. - Incomplete flushing of the SI pumps following SI actuation could also be the cause of the failure. Corrective actions, including the following, were taken. - Monitoring the BIT discharge line boric acid concentration daily. - Venting the SI pumps daily. - Revision of the emergency procedure for recovery from spurious SI. In addition, this report describes that in some plants the removal of BIT or decrease of boron concentration in BIT was permitted because of various operational problems and recent improved calculations for steam line break incidents.

Report # 0572G0 UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE PRIMARY CONTAINMENT.

During full power operation, a failure of electric pressure regulator (EPR) caused an automatic scram. Despite successful scram, one of the two scram discharge volumes (SDVs) did not fully isolate due to failures of two drain valves. As a result, the hot reactor coolant leaked through the control rod drive (CRD) seals, and into the SDV and reactor building equipment drain test (RBEDT). The heat-up of the SDV and steam flashed from the RBEDT caused activation of a portion of the deluge fire system, which threatened the operation of some safety related equipment. The reactor coolant leak into the reactor building resulted in elevated airborne radiation level. Scram reset was necessary for terminating the release. However, by design reactor power must be reduced to 600 psig in order to reset the scram. This pressure reduction could be done by initiation of the isolation condensers (ICs). However, problems including high coolant level delayed initiation of ICs. After the actuation of the electromatic relief valves due to high reactor pressure, the reactor water level decreased and pressure control was initiated. When pressure was reduced to less than 600 psig, the scram was reset and the release of reactor coolant was terminated. The failure of EPR was caused by a sticking servo valve due to impurities in the hydraulic oil. Some of the delay in initiating the ICs was attributed to operator error to control the reactor coolant level. Failure of the two drain valves in SDV were attributed to improper stroke adjustment and a design deficiency. With regard to the failure of SDV drain valves, the possibility of generic cause is pointed out in IRS 572G2 and IRS 572G3.

Report # 058803 LOSS OF ALL IN-PLANT AC POWER, REACTOR TRIP AND WATER HAMMER AT xxxx

The plant was operating at 60 per cent power with one vital bus being provided with off-site power through the "C" auxiliary transformer. This transformer developed an electrical fault resulting in the loss of this vital bus. Control room operators responded by manually scramming the reactor pursuant to procedures for loss of a vital bus. The main generator was then tripped which resulted in the loss of all in-plant AC power. As designed, the diesels started, but did not automatically energise the emergency buses. Operators restored in-plant power from the switchyard in about four minutes. Following the reactor scram, a water hammer occurred associated with the "B" steam generator which is believed responsible for a sizeable feedwater leak that could not be isolated because of the environmental conditions at this location. This leak was from a check valve in the bypass line for the "B" steam generator feedwater regulating valve. Other failures during this event included a ruptured feedwater heater. Cooldown was within technical specification limits and the plant was bought to cold shutdown. The most significant aspect of the event involved the failure of five safety-related check vales in the feed-water system whose failure occurred in less than a year, without detection, and jeopardized the integrity of safety systems. The event involved a number of equipment malfunctions, operator errors, and procedural deficiencies. This report is followed by IRS 0588.02 and IRS 0588.03.

Report # 059400 SYSTEM INTERACTION EVENT RESULTING IN REACTOR SYSTEM SAFETY RELIEF VALVE OPENING FOLLOWING A FIRE PROTECTION DELUGE SYSTEM MALFUNCTION.

An instrument water supply vent valve was damaged by the dragging of a crane hook and the water supply line eventually depressurised, causing a portion of the fire protection deluge system to actuate. Approximately 15 to 25 gallons of water backed up into the ventilation header from charcoal filter housing before the system was secured. Eventually the water sprayed onto the analogue transient trip system (ATTS) panel through a hole in the ventilation piping. The water intrusion into the panel caused the short-circuit in one of two redundant panel power supplies. As a result, low-low-set safety relief valve (LLS-SRV) began to cycle open and thereafter stuck open. The operator manually scrammed the reactor from 25% power. A false turbine high exhaust pressure trip signal was generated, temporarily disabling the high pressure core injection (HPCI) system. The reactor core isolation-cooling (RCIC) system was inoperable at the time, so neither HPCI nor RCIC was available. The water level was restored and maintained by the reactor feedwater system until MSIVs were shut. The back-up was caused by plugged drains in the charcoal filter housing. A procedure was not prepared for cleaning the ventilation plenums and drains in the charcoal filter units. Drains in the remaining filter units were cleaned and inspected. Cleaning and inspection procedures were prepared.

Report # 060100 SHUTDOWN CAUSED BY FAILURE OF NEUTRON OVERPOWER DETECTORS

While returning the unit to service following a maintenance outage, the reactor power was being held at 70% full power and the neutron overpower (NOP) detectors trip test was performed. During the testing it was found that the test circuit of three NOP detectors was not normal. For two of the three failed detectors, spare detectors were available. It was decided to raise reactor power with the suspect detector in service. When reactor power reached 82% full power, the suspect detector signal fell to zero and the operator rejected the trip channel (that is, one of three channels of the shut down system (SDS)). Soon after, a detector of another trip channel failed to zero and an orderly shutdown was initiated. When reactor power was being reduced, a second detector of the latter channel failed and spiked high. This resulted in the firing of the SDS.(Because the rejected channel had been placed in tripped state.) All the NOP detectors were replaced. Some modification of the trip testing method is under investigation.

Report # 061000 FLOODING RESULTS FROM EXPANSION JOINT FAILURE AND INSTALLATION ERROR

While Unit 1 was operating at 85% power, one circulation pump (CWP) tripped due to the rupture of a 9 foot diameter flexible expansion joint between the discharge of the CWP and its discharge valve. Water gushed from the failed joint at about 200 gpm into the lake screen house (LSH). Subsequently the unit was manually scrammed Unit 2 (in cold shutdown) CWPs were shut down because of the high water level in LSH. Plant non-essential service water pumps were de-energised to prevent water damage. Cool down of the reactor was initiated with a combination of the reactor core isolation cooling system and the cycling of the main steam safety release valves (SRVs). The process computer was shut off due to a loss of air conditioning. The reactor water clean-up system was isolated on high differential flow following cycling of SRVs. Containment temperature exceeded the technical specification but the pressure was kept below the isolation pressure by venting. Approximately 675,000 gallons of water leaked into LSH. The fire protection system diesel fire pumps were used to establish the cooling of the station air compressor, turbine lube oil and the drywell pneumatics. The primary cause of the incident was fatigue failure of the discharge valve gear operator mounting bolts. As root causes the following were indicated: - The applied assembly torque for the mounting bolts was significantly less than the specification. - The assumption used for the analysis of the valve operator was inadequate and design margin was presumably insufficient.

Report # 061600 SAFETY INJECTION AND REACTOR TRIP DUE TO LOSS OF STATION INSTRUMENT AIR PRESSURE.

While the unit was at full power, a low station air pressure alarm was received. Although the standby station air compressor started, the reduced air pressure caused one main steam isolation valve to drift closed. The resultant increased steam flow caused steam line pressure to drop in the other two lines. The pressure drops led to safety injection (SI) signal actuation, which resulted in a reactor trip. After the plant was stabilised in hot standby, water was found spraying from both of the two low head SI pump seismic wedge control seals. Both pumps were declared inoperable, which required an entry into cold shutdown. During the incident, one RHR pump failed to start due to a miscalibrated overcurrent protection relay. The low station air pressure was the result of a failed solder fitting on the instrument air system. The solder fitting failure was due to a combination of a faulty heater control and associated repair activities. The leak from both low head SI pumps was determined to be due to aged O-rings and gaskets, which were damaged by a minor flow induced pressure transient. As corrective actions, the failed fitting was brazed along with other affected fittings. A protective step was installed over this line in order to prevent accidental strikings. All the O-rings and gaskets on each pump, were replaced and gasket seals were properly tightened.

Report # 061800 BLACKOUT SIGNAL AND INTERACTION EVENT BETWEEN UNITS

The unit was operating at 95% power. While a Unit 1 nuclear equipment operator (NEO) was performing a routine operability test on the Diesel Generator (DG), he inadvertently opened the breaker that supplied normal incoming power, which caused a blackout signal. The DG sequencer actuated, the bus load shed and then the sequencer allowed all load groups to re-energize. Due to the blackout, the containment chilled water chillers tripped and the containment pressure began to rise. A Unit 2 nuclear control operator (NCO), who was in the process of making up the Unit 2 volume control tank (VCT), came over to Unit 1 to reduce containment pressure by opening containment air release and the addition valves. About 25 minutes later, the chillers were returned to service and the pressure restored. The blackout also caused a Unit 2 VCT outlet valve to close. With flow from the refueling water storage tank still being supplied, level in the Unit 2 VCT continued to rise and the VCT was potentially over pressurized. If the NCO who came over to Unit 2 had been present this situation could have been prevented. Corrective actions, including the following, were taken: - An internal letter was issued to the station stating that control room operators assigned to Unit 2 will under no circumstances be used on Unit 1, or vice versa; - The control circuits of the chemical and volume control systems will be moved to unit related power sources; - The modification to provide a control annunciator on high chiller temperature was performed.

Report # 0630G1 LOSS OF OFFSITE POWER CAUSED BY PROBLEMS IN FIBER OPTICS SYSTEMS

During operation at 52% power, the unit tripped because of a total loss of offsite power. As designed, the emergency diesel generators started and the emergency safety systems actuated. About 25 minutes after the reactor trip, offsite power was restored. Four days later, when the licensee was troubleshooting the equipments as a result of the above described incident, a total loss of offsite power occurred again. The emergency diesel generator and emergency safety feature system worked as designed. About 13 minutes after the reactor trip, offsite power was restored. For both events, the loss of offsite power was caused by the apparent malfunction of a multiplexer in the fiber optics system that controlled breakers in the plant switchyard. The ultimate cause of failures of the fiber optics multiplexer had not been determined yet. Control of the affected switchyard breakers was hardwired and the multiplexer control was bypassed. Refer to IRS 0630.G2

Report # 063100 REACTOR TRIP, POSSIBLY DUE TO SHORT PHOTOHELIC CELL IN OUTPUT POWER SUPPLY, WITH ONE REACTOR TRIP BREAKER FAILING TO OPEN.

During the 80% power operation, the reactor tripped from a spurious indication "Loss of Reactor Coolant Flow" on one loop. A few minutes into the recovery it was observed that one of the two redundant reactor trip breakers had not opened and 2 of the 4 main feedwater isolation valves (FWIV) had not closed. Concerning FWIV, because of the feedwater regulating valves closure, the failure did not pose any operational concerns. Investigations revealed the following: - The intermittent failure of the photohelic cell caused the output voltage of the Control Room Instrumentation Distribution Inverter (CRDI) to decrease momentarily. - This voltage dip caused the spurious indication of loss of reactor coolant flow. - The trip breaker had failed to open because of high friction forces in undervoltage trip assembly device caused by a manufacturing deficiency and degradation of lubricant. - The photohelic cell failure may be related to the troubleshooting work. To prevent recurrence, corrective actions including the following, were taken: - Caution on the importance of failure in equipment, powered from CRDI bus inverters - Thorough inspection of the reactor trip and bypass breakers including the replacement of undervoltage trip assemblies.

Report # 063300 REACTOR TRIP RESULTS FROM ERRONEOUS CONTROL BOARD INFORMATION DUE TO INVERTER FAILURE.

At the unit, one inverter failed. This caused a loss of one main feedwater pump, a plant integrated control

system (ICS) runback to 55% power, tripping of the reactor protection system (RPS) channel, and interruption of the primary power supply to the control rod position indication (RPI) system. A reactor trip had not occurred, since two RPS channels are required to trip. Due to an improperly adjusted backup power supply, the RPI system momentarily indicated that all control rods were on the bottom of the core. This information combined with the alarms from the tripped RPS channel, caused control room operators to believe a reactor trip had occurred. In accordance with plant procedures for a reactor trip, operators manually tripped the main turbine causing an anticipatory reactor trip. During the event there was the smell and sight of smoke in the control room because of the overheating of relay due to sustained low voltage conditions. Following the turbine trip a spurious emergency feedwater initiation and control system actuation of emergency feedwater occurred. The overcooling caused the pressuriser level to decrease; the manual opening of one high pressure injection valve maintained pressuriser level. The main feedwater block valve of another system did not shut automatically because of a wiring failure. The Control Board
Operator manually closed the valve. Corrective actions are as follows: Development of recommendations for verifying the vital bus static switch undervoltage transfer setpoints. In addition, the inverter/bus voltage and current alarms are being reviewed. Readjustment of the current limited power supplies for the CRD system, including the preventive maintenance program to periodically verify the setpoints.

Report # 064100 INADVERTENT OPENING OF A RELIEF VALVE DUE TO A SPURIOUS CONTACT BETWEEN WIRES INSIDE AN ELECTRIC PENETRATION

While at full power, the plant experienced an inadvertent opening of a relief valve, IORV, which was detected, and the relevant procedures undertaken. Attempts to close the safety relief valve involved, only succeeded after replacement of all the fuses, positive and negative side, on the relief and automatic and depressurisation logic. The verification identified a spurious contact between wires inside a module of an electric penetration. As a consequence of the SRV opening, the technical specifications limit of the suppression pool water temperature was exceeded. Perusal of the plant documentation identified a preexisting spurious contact between two further wires in the same module since the previous week, but which had been misinterpreted and erroneously attributed to a ground fault. Insulation verifications showed incipient insulation resistance degradation for about 20 wires. The module was then replaced. An in-depth analysis was started on several aspects such as: the causes of the low insulation resistance, the applicability to other modules, operating actions and diagnostic features. Potential consequences searching for this type of failure showed the possibility of a jammed-open valve due to indirect supply of its pilot solenoid through a short circuit.

Report # 064700 INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION AND SUBSEQUENT REACTOR TRIP

While the reactor was operating at about 50% power during start-up testing, a ventilation fan in one train balance of the plant engineered safety features actuation system (BOP ESFAS) auxiliary relay cabinet failed and caused the cabinet to overheat, resulting in the BOP ESFAS logic card malfunctioning and activation of BOP ESFAS. This caused de-energising the 1E bus and shedding its electric loads. One emergency diesel generator (EDG) started in test mode, so the output breaker did not close. Manual attempts to re-energise the loads on the bus were unsuccessful because of the continuing load shed signal, and during this effort the EDG tripped on reverse power. It was not until the fuses in the BOP ESFAS cabinet were removed, that the bus was re-energised. During this event the essential chiller of another train tripped on low refrigerant temperature causing the train to be unoperable. Because of the loss of both trains of high pressure safety ejection, low pressure safety injection and containment spray, the reactor began to be brought to hot shutdown. However, with the reactor power at about 2%, the control room operator was unable to maintain the feedwater flow to both the steam generator (SG) with the running main feedwater pump (MFWP) because of insufficient pressure difference. The MFWP was manually stopped and the non-essential auxiliary feedwater pump (AFWP) was started. However, because of the malfunction of the control room indication, the non-essential AFWP was tripped and essential AFWP started. This caused the low level of one SG and led to the reactor trip.

Report # 064800 LOSS OF INSTRUMENT AIR CAUSES REACTOR SCRAM WITH HIGH PRESSURE COOLANT INJECTION SUBSEQUENTLY FAILING.

The reactor was operating at 98% power and the back-up instrument air system (IAS) was placed into service due to the trouble shooting of normal IAS. However, because of the high moisture content of air, the air pressure dropped to a level which activated the feedwater flow control valve (FCV) lockup circuits. But one feedwater FCV did not lock, because the valve's positioner malfunctioned. As a result the valve drifted open, causing a high reactor water level which tripped the turbine and reactor. High Pressure Coolant Injection (HPCI) was initiated as designed (feedwater flow shifted to HPCI mode) but was unable to manually reset because of the malfunction of the turbine's governor unit. To control the reactor water level one feedwater pump was locked out and restarted several times. But his action resulted in the failure of the auxiliary oil pump start circuit. Therefore, another feedwater pump was used to maintain the reactor coolant level. The transient analysis indicates that all six electronic relief valves (EMOVs) should have been opened by the pressure spike upon turbine trip and turbine stop valve closure. However, one of the EMOVs failed to open due to a wear-related failure of the solenoid actuation. Inspection of other EMOVs revealed less severe wear in two other valves. Corrective actions including the following were taken: - The failed components including the feedwater FCV positioner, the instrument air dryer and the turbine emergency governor, were repaired. - The failed EMOV was replaced. The actuator assemblies of the other five EMOVs were replaced.

Report # 064900 RECURRENT SPURIOUS CONTAINMENT ISOLATION, MAIN STEAM ISOLATION VALVE FAILURES TO REOPEN AND PRESSURIZATION OF RESIDUAL HEAT REMOVAL SYSTEM

On 4th April 1986, while the reactor shutdown was in progress because of the oil leak in the main turbine control oil system, a Group I primary containment isolation occurred unexpectedly resulting in the reactor scram. At that time, the low pressure coolant injection system was inoperable due to an unrelated problem. The containment isolation signal was promptly reset following the scram. But, the outboard main steam line isolation valves (MSIVs) could not be reopened. As a result, the main condenser was not available as a heat sink. Most of the reactor cooldown was controlled by directing reactor steam to the high pressure coolant injection turbine. On 12th April 1986 a similar incident took place again. Result of the investigation and corrective actions: - concerning unexpected containment isolation, no cause has been identified. But the reactor mode switch failure is under suspicion. - Concerning MSIVs failure to reopen, the cause is that the pilot poppets are detached from the valve stem or inhibited from fully opening so that the differential pressure across the main poppet would prevent the MSIV air cylinder from opening the valve. Pilot poppet set screw did not prevent the poppet from unscrewing from the stem. The design change of MSIVs will be evaluated. - Concerning RHR high pressure, the cause was slow leak of reactor coolant through the check valve and two motor operated injection valves. Pressure monitoring equipment will be installed on the RHR piping and the related maintenance procedure will be revised.

Report # 067600 COMPLETE LOSS OF AUXILIARY FEEDWATER CONTROL FOR BOTH STEAM GENERATORS

Following a turbine trip, a design error in the logic led to a turbine bypass unavailability, a main steam pressure increase, then a reactor scram and the actuation of the safety function for secondary side incidents. During the subsequent course of the transient, the auxiliary feedwater system AFS injection valve at both one-through steam generators OTSG were unintentionally electrically blocked in the closed position after 6 minutes for one train, and after 39 minutes for the other one. The OTSGs were fed by the AFS via the bypass throttling valve. In this way an inventory sufficient for heat removal was maintained. The situation was restored after one hour by manually opening of the AFS injection valves. During the transient pressure oscillations were generated by opening and closing of steam generator relief valve (up to 100 mb) and caused the initiation of "AFS stop" and "AFS reset" signals, at close intervals. In these conditions, the close command has the priority in the drive control device. As counter measures, the transmitters for the SG level measurements were electrically damped. Additionally, the drive control device's open-command is blocked as long as the close command from the memory device is active. A test confirmed the proper behaviour of the AFS inspection valve after these modifications.

Report # 067800 WATER DRIPPING IN CONTROL ROOM.

On February 5, 1985 at 12:26 hours, when the reactor was operating in steady conditions, water was detected on the control room ceiling. The dripping started on the panel area of the electric systems, located in a wing of the control room, and expanded lately to a succession of panels of the Division 1, as are the ADS, RHR, LPCS and RCIC, placed in the other side of the control room. At 12:34 spurious alarms began to appear in ECCS, recirculation system and reactor panels, and at 12:42 manual reactor scram followed. The cause was water flooding in the cables room above the control room. This was due to the handling of the tripping line valve of the Fire Protection System, which provoked the functioning of this system. The water then reached the control room through the penetration of cables whose sealing was not waterproof. The design specification of the sealing was wrong.

Report # 068000 SPURIOUS OPENING OF PRESSURIZER SPRAY VALVE

Due to a mechanical failure, the pneumatic-actuated pressurizer spray valve PCV-400A opened, while the control signal sent from the valve controller (the only indication of the valve position available to the operator) indicated fully closed. The primary pressure started to decrease, despite a power level reduction initiated by the operators, causing finally a reactor trip and safety injection. The depressurization continued until the operating crew was able to identify the valve position by direct means (sending an auxiliary operator to visually check the valve position). Then, the reactor coolant pump was shutdown stopping the flow through the pressurizer spray, the depressurization stopped and the plant was maintained in hot standby condition, using normal operation procedures. The valve was designed to fail safe. This means that the valve should have failed "closed", and this is the case if the failure is caused by loss of pneumatic supply to the actuator or electric signal from the controller. However, in the event of mechanical failures such as the above the valve failed "open". The failure involved rupture of the mechanical link between the valve stem and the pneumatic positioner. Moreover, the failure is such that no manual remote actuation was able to close the valve. It is possible that the failure was due to previous maintenance work but this is not clearly established. The licensee will install, in the next refueling outage, direct indication of the valve position, using limit switches. In addition, the situation of other pneumatic control valves of similar design will be reviewed.

Report # 069300 UNLABELED SWITCH RESULTS IN INADVERTENT ACTUATION OF DELUGE SPRAY SYSTEM AND SUBSEQUENT SCRAM

While in start-up operation, a fire protection curtain (deluge spray) in the auxiliary building was inadvertently actuated by a construction employee. The actuation occurred when the employee attempted to open a remote data acquisition cabinet (RDAC) panel to seal a conduit, and mistook a solenoid actuation switch for a door latch. The water curtain ran for about 20 minutes and overloaded the floor drain system. Water backed up on the floor and ran into the bottom of two motor control centres (MCCs) that are mounted directly on the floor. The water then ran down unsealed penetrations inside the MCCs onto the top of heating, ventilation, and air conditioning (HVAC) ductwork underneath the floor. The water ran along the top of the HVAC ductwork and cascaded onto the top of and into a load center. This caused a short which the load center transformer saw as a high current demand. This caused the transformer to burn up and the associated supply breaker to trip. The supply breaker also fed 2 nonsafety-related load centres. The hydraulic pump which supplied the turbine bypass valves tripped and caused the bypass valves to fail closed. As the main steam line drains were opened to reduce the increasing reactor pressure, and rods were being inserted, the intermediate range monitors (IRMs) were down ranged to maintain on scale readings. Subsequent level changes and a sudden cold feedwater injection caused a rapid increase in IRM readings leading to the scram. Investigations revealed that the solenoid actuation switch nametag was missing.

Report # 074700 LOSS OF OFF-SITE POWER DUE TO UNNEEDED ACTUATION OF STARTUP TRANSFORMER PROTECTIVE DIFFERENTIAL DELAY

The unit was operating at approximately 80% power. During racking out the emergency diesel generator (EDG) B breaker for modifications, the emergency bus 2 (E2) lost power, leading to a reactor trip. Subsequently, the main generator oil circuit breaker opened and plant auxiliary loads, including E1 and E2, shifted to the start-up transformer. But, approximately 1 second later, a West bus lockout occurred in the 115kU switchyard which de-energized the start-up transformer, resulting in a loss of off-site power to auxiliary loads and emergency buses E1 and E2. EDG A automatically started. When the loss of off-site power occurred, all the reactor coolant pumps tripped and the plant was cooled by natural circulation. The plant was stabilized at hot shutdown conditions with operating steam generator power operated relief valves. It is speculated that the cause of the initial loss of E2 was a blown fuse or a loose fuse holder. The cause of the loss of off-site power was the operation of one phase differential relay associated with the start-up transformer during the transfer of auxiliary loads and emergency buses. The cause of the phase differential relay actuation was the direct current saturation of the current transformer on the primary side of the start-up transformer.

Report # 0785G0 OPERATIONAL EXPERIENCE INVOLVING LOSSES OF ELECTRICAL INVERTERS

Inverters in nuclear power plants provide "uninterruptible" vital AC electrical power to safety- and non-safety-related instrumentation and control systems. Generally, loss of this function results in some type of undesirable system condition and/or plant transient, including unnecessary actuation of safety systems such as reactor protection and engineered safeguards systems; loss of indicators that provide plant status information; system disturbances, including reactor coolant system transients; improper response of the feedwater and steam generator water level control systems; loss of safety-related electrical equipment functions; damage to mechanical equipment; and challenges to operators and the remaining functional equipment. Such conditions and-or transients clearly have significant safety implications since they result in challenges to safety equipments and plant operations and-or a degradation of plant equipment. A total of 142 events involving inverter losses occurred at U.S. nuclear power plants during 1982 through 1984. These 142 events took place at 51 distinct plants: 26 PWRs designed by Westinghouse, 11 BWRs by General Electric, 9 PWRs by Combustion Engineering, 4 PWRs by Babcock and Wilcox, and 1 HTGR by General Atomic. Three potential failure mechanisms for inverters involved relatively high ambient temperature and-or humidity within inverter enclosures; interconnecting and physical arrangements for the inverter circuit components; voltage spikes and perturbations. Recommendations are made to minimize the number of inverter loss events.

Report # 0819G0 DEPRESSURIZATION OF REACTOR COOLANT SYSTEM IN PRESSURIZED WATER REACTORS

This report addressees potential problems resulting from the loss of pressure control in the reactor coolant system (RCS) which could affect the operator's ability to depressurize the reactor coolant system in a timely manner during a steam generator tube rupture accident, or to control the plant during natural circulation cooldown. Two events are addressed here which demonstrate the importance of maintaining the capability to depressurize the RCS in emergencies. At xxxx, a double ended rupture of a single tube occurred in the "C" steam generator causing an initial break flow of around 600 gpm. The operators were able to minimize the radiological release during this event because they were able to control primary pressure. None of the secondary atmospheric relief valves were actuated in this event. A reactor trip occurred at xxxx Unit 2 when a technician inadvertently grounded a 120V AC instrument bus, causing a spurious loss-of-reactor-coolant-pump reactor trip signal. The operators secured the reactor coolant pumps after 5 minutes of operation because component cooling water was not available to cool the motor bearings and the thermal barrier. The high-pressure safety injection pumps continued to operate after the receipt of the safety injection signal. The resulting rise in reactor coolant system pressure caused a power-operated relief valve (PORV) to lift numerous times. Normal pressurizer spray was not available to control the primary system pressure rise once the reactor coolant pumps were tripped. These events show that the PORVs and the auxiliary pressurizer spray system can be important to safe shutdown of the plant under certain emergency conditions.

Report # 086100 FALSE TRIGGERING OF REACTOR PROTECTION SIGNALS DUE TO A FAILED CLOCK GENERATOR

As a result of a false initiation of output signals in the reactor protection system, a series of reactor protection measures, including trip of a reactor coolant pump and closing of all full load feedwater valves, was initiated. The reactor subsequently tripped due to low water level in steam generator 1. The supply to this SG was restored by opening the shutoff valves manually. The false initiation was recognized early in the event, but it took some time to identify its source. The cause of the false signal was an electrical short circuit between the outputs of a clock generator due to a too small distance combined with a conducting layer of dirt between two transistor housings. Because of the pivotal role of the clock generator, its failure resulted in a multitude of malfunctions of safety features. To prevent a recurrence it was recommended that a clock generator diagnostic device be installed and that all plants be reviewed to determine whether a failure in the clock generator system could affect a complete safety function.

Report # 086300 DAMAGE OF AN INSTRUMENT TRANSFORMER CAUSES LOSS OF OFF-SITE POWER TO BOTH UNITS

A loss of off-site power occurred in both units of xxxx NPP as a result of the damage of an instrument transformer in the high voltage circuit of the 220 kV generator transformer of unit B. Consequential breaker actions and sparking caused by the instrument transformer damage lead the plants to the loss of off-site power. The emergency diesels of both units started properly. After approximately half an hour the emergency power supply could be replaced by supply via the reserve auxiliary transformer after reconnecting this transformer to the 220 kV grid. After inspection of the 380 kV line the 380 kV power supply to unit B was also restored. The root cause of the incident is under investigation.

Report # 087400 FAILURE OF TWO SETS OF REDUNDANT PRIMARY CONTAINMENT ISOLATION VALVES AT xxxx UNIT 2

On January 2, 1988, while xxxx Unit 2 was at 69% power, a failure of four primary containment isolation valves (PCIVs) occurred. While preparing for a planned shutdown, the operator began an orderly power reduction by a combination of reduced circulation flow and control rod insertion: difficulties were encountered in maintaining the condenser vacuum. Following a manual scram, the reactor vessel level dropped below the low level setpoint which sent an automatic closure signal to the four PCIVs. The equipment drain PCIVs failed to close on an automatic signal and the floor drain PCIVs failed to close on a subsequent manual signal. Three valves had closed after six minutes, which provided isolation of the breach of primary containment. All four valves ultimately were found closed after eight minutes since the generation of the original closure signal. The simultaneous failure of both of the redundant drywell equipment drain and drywell floor drain PCIVs occurred due to multiple and diverse causes such as solenoid sticking and relay contact failure. This occurrence was a previously unanalyzed condition. Under certain accident conditions, the simultaneous failure of all four redundant PCIVs could have resulted in a direct, unmonitored and unisolable path of release of radioactivity to the environment. Under accident conditions any attempts of local closure would have been impeded due to persistEnt high radiation fields. The licensee revised operating and training procedures such that these valves will be normally closed during plant operations except for periodic pumpout of collected liquids.

Report # 087700 LOSS OF RECIRCULATION PUMPS ACCOMPANIED BY SEVERE POWER OSCILLATIONS AT xxxx UNIT 2

On March 9, xxxx Unit 2 underwent a dual recirculation pump trip following which the unit experienced excessive neutron flux oscillations while it was in natural recirculation. The unit had been operating at 84 percent power when an instrument technician performing surveillance on the reactor core isolation cooling (RCIC) initiation logic, opened a wrong valve. The resulting hydraulic perturbations on the anticipated transient without scram (ATWS) switches caused the recirculation pumps to trip. The loss of recirculation pumps caused a power reduction to about 40 percent power. The loss of these pumps caused a perturbation in the feedwater heater system which caused the isolation of some feedwater heaters causing the loss of some preheating of the feedwater. The addition of cooler feedwater to the reactor caused instabilities resulting in power surges and the reactor eventually scrammed on high neutron flux. The plant was taken to cold shutdown. The NRC dispatched an Augmented Inspection Team (AIT) to the site. Based on the licensee investigation and the NRC review, it was concluded that the event had significant generic implications. The licensee revised its operating procedures, instructed plant operators in responding to such events, and re-evaluated engineering analyses. The NRC issued communications on this event urging the licensees and construction permit holders of the boiling water reactors to ensure that adequate operating procedures and instrumentation are available, and adequate operator training is provided to prevent uncontrolled power oscillations during all modes of reactor operations.

Report # 088200 SHORT CIRCUIT IN MAIN TRANSFORMER CAUSING REACTOR TRIP, ELECTRICAL FAILURES AND SECONDARY SIDE TRANSIENTS

While operating at full power, a failing insulator of the main transformer, causing single phase earthing, actuated its Buchholz relay and resulted in turbine and reactor trips. Transferring power from the house load transformer to one of the two start-up (auxiliary) transformers failed because its differential current protection relay disconnected all normal and emergency 6 kV busbars from the grid. The resulting emergency power situation was characterized by several anomalies, i.e. failure of the emergency cooling water system, vapour binding of the emergency feedwater pumps, instabilities in the deaerator/feedwater drum pressure and an incorrectly carried out resynchronization resulting in a short loss of one emergency power busbar. It was determined afterwards that procedural deficiency and remedy action taken played an important role in these anomalies. Repairs and subsequent testing of the main transformer lasted 12 days and the plant had a forced outage of 15 days.

Report # 088400 REACTOR TRIP, FOLLOWED BY AN ECCS ACTUATION, DURING A QUALIFICATION TEST OF A MODIFICATION ON A SECOND LEVEL PROTECTION DIESEL GENERATOR

During a qualification test of a modification on a second level protection diesel generator, the electric power of a 220 V AC busbar, feeding one second level protection instrumentation train, was lost. Due to an improper electric refeeding of this instrumentation, an atmospheric relief valve on a main steam line opened together with a trip of a primary coolant circulation pump. As a consequence, a reactor trip occurred, followed after 10 minutes by an ECCS actuation due to low primary coolant pressure. The cause of the event was a human error: the operator didn't use the written instruction and made some operation in a wrong sequence.

Report # 089700 REACTOR TRIP DUE TO FAILURE OF REGULATING SYSTEM IN-CORE FLUX DETECTORS

A loss of reactor regulation occurred due to degradation of an in-core flux detector. The regulating system raised both the local and gross power enough that protective system ion chamber signals caused an automatic reactor trip. Subsequent analysis demonstrated that, even in the absence of the protective system trip, the power transient would have been limited and fuel integrity would have been maintained. A similar event occurred again several days later, after which time the faulty flux detector was repaired. It is necessary to periodically reinforce to operating staff the requirement for objective review of readily available information to ensure correct diagnosis of events.

Report # 091400 FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS

This report addresses a potentially generic problem involving flow blockage in safety-related piping interconnections due to biofouling. This condition may not be detected in a timely manner due to stagnant water in system interconnecting piping which is not routinely flushed or flow tested. On March 9, 1988, xxxx Unit 2 was at 20 percent power. As the feedwater regulating valves were being placed in automatic, the B steam generator (SG) feedwater regulating valve (FRV) failed open. This caused a feedwater swing and after the operators assumed a manual control of the B-SG FRV, a high-high level in the D-SG occurred which resulted in a main turbine trip, a main feedwater isolation, and a main feedwater pump (MFWP) trip. The motor-driven auxiliary feedwater pumps (MDAFWPs) started upon the loss of the MFWPs. The level in SGs A and B, however, kept dropping, and the reactor tripped on low-low SG level setpoint. The initial auxiliary feedwater flow (AFW) had been normal and later began to drop. The SG-A flow control valve (FCV) was fully open. When the SG-A and SG-B FCVs were disassembled, they were found to have been clogged with Asiatic clam shells. The licensee initiated various preventive programs. This event highlights the importance of maintaining these lines free of clams, corrosion and other foreign material.

Report # 093900 TRIP OF REACTOR 2 DUE TO LOSS OF 11 KV UNIT BOARD

An 11 kV unit board 2B tripped and reactor 2 tripped due to loss of supplies to channels 3 and 4; reactor 1 remained on load and was not significantly affected. The cause of the 11 kV board fault was probably the combination of water ingress and a foreign body.

Report # 095800 PARTIAL BLOCKAGE OF THE WATER INTAKE OF ONE UNIT, AND LOSS OF OFF-SITE POWER TO THE TWIN DURING COLD WEATHER

The incident which occurred at the xxxx Unit "A" natural uranium, graphite-gas power station, on 12 January 1987 was two-fold: a) a partial blockage of unit 1 water intake caused a reactor trip and a one-hour unavailability of all the unit's auxiliary turbo-generators; and b) two hours later, a voltage reduction on the transmission grid caused by the trip of xxxx thermal station, led to the trip of the Unit 2 two main turbo-generators, two line breakers and reactor. The root cause of this two-fold incident is the cold weather which prevailed over the west of France: ice floes, transported by the river Loire caused the Unit A-1 water intake blockage; the cold weather was also partially responsible for the trip of the xxxx thermal station. This incident highlighted the risks associated with cold weather conditions, and reinforced the decision that had already been taken concerning the importance of installing independent diesel generators at that station, and the importance of instituting procedures addressing a general loss of electrical supplies or a loss of raw cooling water. Furthermore, this incident shed light on an inadequate configuration of the pumping station and on errors in the general station operating instructions which were corrected.

Report # 102502 FIRE IN ONE TURBINE GENERATOR GROUP AND SUBSEQUENT FAILURE OF SAFETY SYSTEMS

On October 19th, as a consequence of a mechanical failure in one of the turbogroups, a leak of lubrication oil together with h2 from the generator cooling system, started a large fire which lasted for 6 hrs. This fire caused several systems failures which induced among other things, a significant flooding of reactor building (80 cm high). As a consequence core cooling was compromised, due to the loss of two gas circulators and the regulation problems of feedwater to the main heat exchanger. The fire was extinguished 6 hrs after initiation of the event, water extraction from reactor building was ended 12 hrs later and feedwater to main heat exchanger was stabilized 19 hrs after event initiation. Cooling limits were not exceeded during the event and there was no radioactive release or exposure above limits. The plant will remain shutdown until the implementation of significant design modifications and other recommendations derived from this event.

Report # 103500 REACTOR TRIP DUE TO UNDERVOLTAGE ON CONTROL ROOM INSTRUMENTATION DISTRIBUTION PANEL AT xxxx UNIT 2

This IRS report addresses a recent event at xxxx Unit 2. On August 14, 1989, the reactor tripped from 100 percent power as a result of a severe undervoltage on 120 VAC vital bus to control room instrumentation distribution (CRID) panel IV. The undervoltage condition was caused by a silicon control rectifier (SCR) failure in the associated inverter. As a result of the undervoltage, all four wide range steam generator level indicators, the condenser steam dumps, the solid state protection system (SSPS) train "B" output relays, and other instrumentation were lost. Additional unrelated failures, the smell of excessive heating of unfused CRID-IV components and extensive relay chattering complicated recovery from the event. The NRC dispatched an Augmented Inspection Team (AIT) to develop and validate the sequence of events for determination of the adequacy of equipment and operator response. No significant anomalies were identified in either equipment or operator response. The team determined that no plant safety parameters were exceeded during the course of the event, the procedures were adequate and the licensee follow-up actions were appropriate. Although the overall safety significance of this event was determined to be rather low, in the absence of timely and appropriate operator actions, the consequences of the event could have been significant. This IRS report presents the excerpts from the AIT report.

Report # 1046G0 ELECTRICAL BUS BAR FAILURES

This IRS report addresses potential problems resulting from the failure of electrical bus bars caused by cracked insulation and moisture or debris buildup in bus bar housing. Insulation failure, along with moisture or debris, provided undesired phase-to-phase or phase-to-ground faults which resulted in catastrophic failures of buses. Various events at xxxx, xxxx, xxxx, xxxx and xxxx are discussed.
In the events addressed here, the bus bar failures resulted in reactor trips and/or engineered safety feature actuations. Cracked insulation, resulting from high ambient temperatures and/or contamination from the use of "black" bus bar joint compound, has been observed on bus bars manufactured by Westinghouse, General Electric, and the Calvert company, where Noryl flame retardant insulation was utilized. The respective licensees took various corrective actions. They are: (1) Replacing the damaged bus bar sections with bus bar that were covered with insulation of a different type; (2) substituting "yellow" bus bar joint compound for the "black" joint compound previously used; (3) modifying bus bar enclosures to restrict ingress and accumulation of water and debris; and, (4) instituting enhanced periodic inspections and cleaning of bus bars and their housings. The NRC issued Information Notice 89-64 to alert the licensees of U.S. nuclear power plants to these concerns.

Report # 108802 LOSS OF VITAL AC POWER WITH SUBSEQUENT REACTOR COOLANT SYSTEM HEAT-UP AT xxxx UNIT 1

This preliminary IRS report addresses a recent event of loss of vital AC power and subsequent reactor coolant heat-up at xxxx Unit 1 which had been in a refueling outage for about 4 weeks. The NRC dispatched an incident investigation team (IIT) to the site. This report will be updated after formal publication of the IIT report. On March 20, 1990, xxxx Unit 1 was in mid-loop operation during a cold shutdown. The reactor coolant system (RCS) temperature was 90 F. Unit 2 was operating at 100 percent. Various equipment had been tagged out of service. At about 9:20 A.M., a truck carrying fuel and lubricants in the plant's low-voltage switchyard backed into the support column for the feeder line supplying power to Unit 1 "A" reserve auxiliary transformer (RAT) and the Unit 2 "B" RAT. The insulator for the "C" phase of the feeder line fractured and initiated a phase-to-ground electrical fault which resulted in a loss of power to the Unit 1 "A" RAT and the Unit 2 "B" RAT. The Unit 2 "B" emergency diesel generator (EDG) started and loaded to the de-energized Unit 2 "B" vital bus. However, an improperly connected differential current transformer caused the turbine to trip, and subsequently the reactor tripped. Unit 1 "B" EDG was out of service. The "A" EDG started automatically and shut down in about a minute. A "Site Emergency" was declared at 9:56 A.M. as loss of all on-site and off-site power had lasted more than 15 minutes. Eventually, Unit 1 "A" EDG was manually started, by bypassing some trip features; associated buses were reenergized; and the residual heat removal pumps re-started. The site area emergency was down-graded to an "alert". The RCS temperature rose to 136 F. at a rate of increase of about 1.3 F/minute. If the emergency power were not restored, boiling was expected to occur in about 1 hour and 36 minutes from the beginning of the event. This incident emphasizes the need for carefully assessing the consequences of performing parallel maintenance on…

Report # 108803 LOSS OF VITAL AC POWER AND THE RESIDUAL HEAT REMOVAL SYSTEM DURING MID-LOOP OPERATION AT xxxx UNIT 1 (FINAL REPORT)

This updated IRS report presents the NRC's Incident Investigation Team's (IIT's) assessment of a recent loss-of-vital-AC-power event at xxxx 1. On March 20, 1990, xxxx Unit 1 experienced a loss of all safety (vital) AC power. The plant was in cold shutdown with reactor coolant level lowered to "mid-loop" for various maintenance tasks. Both the containment building personnel hatch and equipment hatch were open. One emergency diesel generator (EDG) and one reserve auxiliary transformer (RAT) were out of service for maintenance, with the remaining RAT supplying both Unit 1 safety buses. At about 9:20 A.M., A truck in the low-voltage switchyard backed into the support column for an off-site power feed to the RAT which was supplying safety power. The insulator broke, a phase-to-ground fault occurred, and the feeder breakers for the safety buses opened. The Unit 1 operable EDG started automatically because of the under-voltage condition on the safety bus, but tripped after about 1 minute. Nearly 20 minutes later, the EDG load sequence was re-set, causing the EDG to start a second time. The EDG operated for about a minute, and tripped off. The EDG was re-started in the manual emergency mode 36 minutes after the loss of power. The EDG remained on line and provided power to safety bus, and the residual heat removal pumps re-started. During the 36 minutes following the loss of safety bus power, the reactor coolant system temperature rose from about 90 F to 136 F. The IIT identified several potential generic lessons, and concluded that the plant staff's generally effective response to this incident compensated for the weaknesses in their procedures. The Unit 2 reactor, which was at 100 percent power when this event occurred, tripped following a turbine trip; the report presents the associated sequence.

Report # 109500 LOSS OF OFF-SITE POWER WITH MULTIPLE EQUIPMENT FAILURES AT xxxxx UNIT 2

This IRS report addresses an NRC Augmented Inspection Team's (AIT's) review of a loss of off-site power (LOOP) at xxxx Unit 2. On January 16, 1990, while Unit 2 reactor was at 100 percent power, the "2D" condensate-booster pump failed due to an internal motor fault precipitating an automatic trip of the feedwater pumps on low suction pressure. This resulted in reduced reactor water level, and ultimately scrammed the reactor. The reserve auxiliary transformer (RAT) tripped on actuation of the sudden pressure relay during automatic transfer of house loads, resulting in interruption of normal AC power. Subsequently, anticipating the loss of the unit auxiliary transformer, the operators manually started the Unit 2 emergency diesel generator (EDG). This action, though well-intended, could have had serious consequences. If the EDG were to be loaded prior to the LOOP with the cross-tie to the normal buses closed (as was the case in this event), then upon LOOP, the EDG will pick up all the non-vital loads. Given this scenario, the EDG design capacity would have been exceeded, if there were an emergency core cooling system actuation. Besides the "2D" condensate-booster pump failure, other problems included failure of the standby booster pump to start, spurious closure of one of the Unit 2 main steam isolation valves, failure of a control switch/indicator, failure of the main generator output breakers to open automatically on reverse power, and the failure of the shutdown cooling pump outlet valve. The AIT concluded that design weaknesses and lack of timely implementation of the BOP preventive maintenance program contributed to the equipment failures. The licensee replaced the RAT and took other corrective actions.

Report # 1109G0 POTENTIAL FAILURES OF ROSEMOUNT PRESSURE AND DIFFERENTIAL PRESSURE TRANSMITTERS DUE TO LOSS OF FILL-OIL

This IRS report addresses a potential safety generic concern associated with failures of Rosemount pressure and differential pressure transmitters, particularly, Model 1153 series B and D, and Model 1154. To date, over 90 failures of transmitters from these series have been reported. In some cases, the transmitter failure was not detectable during operation. However, some of the symptoms that occurred during operation and before failure include the following: Slow drift in either direction; lack of response over the transmitter's full range; increase in the transmitter's time response; deviation from the normal signal fluctuations; decrease in the detectable noise level; deviation of signals from one channel compared with redundant channels; "one-sided" signal noise; and, slow response to a transient or inability to follow a transient. The cause of the failure was confirmed by Rosemount to be a design deficiency causing gradual loss of fill-oil from the transmitter's sealed sensing module. Common cause failures of redundant transmitters may compromise the plant safety during a transient by lack of auto-actuation of safety system or equipment, and/or delayed operator response. In April 1989, NRC issued Information Notice 89-42, "Failure of Rosemount Model 1153 and 1154 Transmitters," to alert the licensees of U.S. nuclear power plants to the failures of these transmitters. Based on the generic implications of the Rosemount transmitter failures, on March 9, 1990, the NRC issued Bulletin 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount". This bulletin was intended to alert the licensees and construction permit holders for U.S. nuclear power plants to promptly provide information on uses, manufacturing lots and failure symptoms etc., of Rosemount transmitters, and to take appropriate corrective actions. A copy of the bulletin is provided as Appendix A.

Report # 115200 REACTOR MANUAL SHUTDOWN DUE TO MISINDICATION OF PRIMARY LOOP RECIRCULATION PUMP MOTOR LOWER BEARING OIL LEVEL

During adjustment operation in its 6th periodical inspection, "Primary Loop Recirculation (PLR) Pump (B) Motor Bearing Oil Level-high" alarm occurred. As the result of confirmation of rated parameters such as motor bearing temperature or vibration, etc. Neither abnormality on the operating conditions of PLR-B, nor specific troubles were observed in the alarm circuit. Reactor was manually shutdown for detailed inspection of the lower bearing of the pump-motor. The result of detailed inspection of the oil level detector was that two reed switch blades for high alarm adhered to each other. It was estimated that this trouble was caused by a deficiency in plating or in dust removal, with gap narrowing due to improper manufacturing; the results caused the contacts to adhere. Following countermeasures were taken: (1) The affected oil level detector was replaced with new one, and similar type detectors were also inspected. (2) It was decided that the integrity of the reed switch in the reed switch type oil level detector of the PLR pump motor is to be confirmed at every periodic inspection. (3) It was decided to perform a test to confirm the adequate reed switch gap clearance at its manufacturing stage.

Report # 1180G0 SOLENOID-OPERATED VALVE PROBLEMS AT U.S. POWER REACTORS - OPERATING EXPERIENCE FEEDBACK REPORT

This IRS report presents the NRC's evaluation of U.S. light water reactor experience (1984-1989) with solenoid-operated valves (SOVs). The NRC study (NUREG-1275, Vol. 6, dated February 1991) focuses on the vulnerability of safety-related equipment to common-mode failures or degradations of SOVs. Information is presented on over 20 representative events in which common-mode failures or degradations affected, or had the potential to affect, multiple safety systems or multiple trains of individual safety systems. While plant safety analyses may not have addressed such common-mode failures or degradations, U.S. operating experience indicates that they are continuing to occur. Furthermore, many SOV failures and degradations observed during the course of this study were beyond the conditions analyzed in the plants' safety analysis reports and are not modeled in present-day probabilistic risk assessments. Common-mode SOV failures reported herein have compromised important safety systems such as emergency AC power, auxiliary feedwater, high-pressure coolant injection, and scram systems, resulting in reductions in safety margins. The events involving common-mode failures of SOVs in which safety systems were affected are considered as precursors to more significant events. The report addresses the root causes of SOV failures and degradations, and recommends various measures to be implemented by the licensees to prevent common-mode failures of the SOVs in safety-related applications. In addition, this study recommends that the U.S. nuclear industry take actions to improve the mechanism for communicating SOV failure data to manufacturers for early detection and resolution of potential generic problems.

Report # 120600 REACTOR SCRAM FOLLOWING LOSS OF FIVE UNINTERRUPTIBLE POWER SUPPLIES AND PARTIAL LOSS OF CONTROL ROOM INSTRUMENTATION (PRELIMINARY REPORT)

This preliminary IRS report addresses an event at 0600 hours on August 13, 1991, involving a reactor trip from 100 percent power at xxxx Unit 2 following a loss of control room annunciators and partial loss of the reactor system and balance-of-plant (BOP) instrumentation. At the time of the incident, the Unit 1 reactor was at 97 percent power. The licensee declared a site area emergency. The NRC dispatched an augmented inspection team (AIT) to the xxxx site to review the circumstances leading to the event and the licensee's actions during and subsequent to the incident. An updated IRS report will be issued after the AIT's event assessment is complete. The loss of control room annunciators and partial loss of the reactor system and BOP instrumentation resulted from concurrent loss of five non-class 1E uninterruptible power supplies (UPSs). At 0622 hours, alternate backup power was restored to the annunciators and instrumentation, and safe shutdown conditions were verified. During the course of the event, the plant operators always had reactor pressure and vessel level indications. There was no core uncovery. All safety systems remained available during this event, except for one train of low-pressure coolant injection which was out of service for maintenance. At 0706 hours, a normal cooldown of the plant was initiated using secondary systems. Offsite power remained available throughout the event. The licensee terminated the site area emergency at about 1943 hours. The potential cause of the loss of the UPSs is believed to be an electrical transient as a result of a fault on one phase of the main transformer. No releases of radioactivity occurred.

Report # 120602 REACTOR SCRAM FOLLOWING LOSS OF FIVE UNINTERRUPTIBLE POWER SUPPLIES AND PARTIAL LOSS OF CONTROL ROOM INSTRUMENTATION (FIRST UPDATE)

This IRS report is the first update to an IRS report by the same title, dated August 14, 1991. That report discussed an August 13, 1991 event involving a reactor trip at 0600 hours from 100 percent power at xxxx Unit 2 following a loss of control room annunciators and partial loss of the reactor system and balance-of-plant (BOP) instrumentation. A preliminary sequence of events has been developed by the NRC's incident investigation team (IIT) and is appended to this update. The team's current understanding of the event is as follows: Phase "B" of the main transformer failed, causing an electrical disturbance in the ac distribution system. This resulted in automatic generator trip, turbine trip and reactor trip. A simultaneous loss of five uninterruptible power supplies (UPSs) caused a loss of several non-safety-related functions, which included reactor control rod position indication, partial loss of reactor power and level indication, control room annunciators, plant communication systems, the plant computer and lighting at some locations. All safety-related equipment and instruments remained operable, with the exception of two trains of low-pressure coolant injection, which had been removed from service for maintenance. Plant operators used the operable equipment and indications to determine the plant status and to ensure that the reactor was shut down and maintained in a safe condition. Power was restored in 30 minutes. A normal cool-down was conducted. Cold shutdown was achieved at 1845 hours and the licensee exited the site area emergency at 1943 hours.

Report # 1326G0 RECENT LOSS OR DEGRADATION OF SERVICE WATER SYSTEM

This report is issued to describe recent operating experience problems involving the loss or potential loss of safety-related heat transfer capability in service water systems (SWSs). Two plants (xxxx, Unit 1, and xxxx, Unit 2) experienced a total loss of their SWSs for short periods, and two plants (xxxx and xxxx Unit 2) experienced significant SWS degradation for short periods.

The xxxx, Unit 1 event shows that personnel error and failure to follow procedures can cause the safety-related SWS to become inoperable. The other three events are examples in which intake debris, caused by adverse environmental conditions, together with personnel errors either caused or could have caused the safety-related SWS to become inoperable. All four events illustrate that recovery strongly depends on human action, particularly with respect to following procedures and accurately communicating information.

NRC Information Notice 92-49 was issued on July 2, 1992, regarding this concern.

Report # 136300 REACTOR SCRAM DUE TO LOW PRESSURE SIGNAL OF THE INSTRUMENT AIR

After noticing that the programmer that automatically shifts the instrument air drying towers was not working properly, the control room operator tried to manually place in service the drying tower which had previously been in stand-by. When performing this operation, the drying tower directional outlet valve failed to open, thus causing low instrument air pressure and, as a consequence, a reactor trip.

The improper operation of the drying tower shift programmer was due to wear of its motor speed reduction device gears.

The directional outlet valve failure to open was due to its seizing, caused by the buildup of a layer made up of wet drying agent entrained by the instrument air coming from the other drying tower, along with rusting of the valve due to the moisture in the air.

The need for more reliable instrumentation and more complete surveillance of the system performance were the lessons learned from this event.

Report # 1446G0 INADEQUATE MAINTENANCE OF UNINTERRUPTIBLE POWER SUPPLIES AND INVERTERS

This IRS report discusses potential failures of uninterruptible power supplies (UPSs) and inverters because of inadequate maintenance. At xxxx Nuclear Station, Unit 2, on August 13, 1991, there was a loss of power from five UPSs which powered the main control annunciator system and other systems important to safety. The failures occurred because of inadequate maintenance of the batteries which supplied power to the control logic. At the same plant on March 26, 1992, a non-safety-related power supply failed to provide power to a radiation monitoring cabinet which in turn caused several engineered safety features to be actuated. The failure occurred because of inadequate maintenance of the power supply output breakers. At xxxx Steam Electric Station Unit 1, a relay in the switchyard was operated incorrectly, which caused the loss of one of the off-site power sources on July 31, 1991. The loss of this power source, coupled with the failure of one UPS to transfer to its backup supply, resulted in a loss of power to some ac instrument panels. The cause of the power supply failure was three failed cells in the power supply battery bank. NRC issued Information Notice 94-24, "Inadequate Maintenance of Uninterruptible Power Supplies and Inverters," on March 24, 1994, to inform licensees of the problem. NRC also issued Incident Investigation Team report NUREG-1455, Augmented Inspection Team report 50-410/91-81, and IRS reports 1206.00, 1206.02, 1206.03, and 1206.04 on the UPS failures at xxxx Nuclear Station, Unit 1.

Report # 146204 REACTOR COOLANT LEAKAGE AND EMERGENCY COOLANT INJECTION DUE TO A RELIEF VALVE FAILURE AND PIPE BREAK

This report is an update that includes the results of a more in-depth root cause investigation.

On December 10, 1994, the reactor was at full power when a liquid relief valve (RV) in the reactor coolant system failed open. The subsequent transfer of coolant inventory into the bleed condenser resulted initially in a reactor setback (a power reduction of 0.5% full power (FP)/sec) on high bleed condenser level, followed soon after by a reactor scram (trip) on low reactor coolant pressure.

Within 6 minutes of the liquid RV failing open, the D2O feed system filled the voids with liquid and re-pressurization of the reactor coolant system occurred. However, the pressure rose too quickly and reached the bleed condenser RVs' setpoint. The two RVs lifted and discharged reactor coolant to the reactor building sump as per design.

After the initial opening, one of the RVs shut and began to chatter. As a result, the valve was damaged and an elbow on the valve piping cracked. This led to a coolant discharge of approximately 63 kg/sec. Also, the vibrations induced by the RV operation caused the copper instrument air lines to fail on all four of the reactor coolant liquid RV actuators, and the remaining three valves also failed open. The reactor coolant leaked from the cracked pipe and into the reactor building sump. The loss of reactor coolant pressure and a rise in the reactor building pressure due to leaking reactor coolant caused the emergency cooling system to actuate. The emergency cooling provided adequate fuel cooling until the normal system shutdown cooling was placed into service. Approximately 140 Mg of emergency coolant light water was pumped into the unit. The total reactor coolant inventory is about 117 Mg.

There was no evidence of fuel failures or damage to the shutdown the steam generator tubes.

The event was rated 2 on the NES scale.

A preliminary investigation and a more in-depth root cause investigation has identified: less than adequate design of the configuration of the bleed condenser RV and its associated piping and embrittlement of the actuator diaphragm, due to thermal aging, on one of the reactor coolant liquid RVs as the major causes of the event. The corrective actions recommended include: 1) implementation of an improved bleed condenser overpressure relief system; 2) review of other RVs in the reactor coolant and connected auxiliary systems to ensure similar inadequate configurations are not present; 3) replacing existing reactor coolant liquid RV actuator diaphragms with new diaphragms of recent cure date (on Units 1-4); and 4) ensure replacement frequency for diaphragms is adequate and appropriate.

Report # 146400 INTERRUPTION OF 48V DC CLASS I POWER RESULTS IN A LOSS OF UNIT LOW PRESSURE SERVICE WATER (LPSW) AND A PARTIAL LOSS OF CLASS IV POWER

xxxx Nuclear Generating Station A, Unit 4, was operating at 60% of full power with 48V dc distribution panel ground fault tracing activities in progress, when both converters, which supply the panel failed simultaneously. The resulting loss of 48V dc control power spuriously tripped the turbine-generator and all of the low-pressure service water pumps. It also disabled a 13.8KV automatic power supply transfer, which left half of the unit without class IV electrical power. The reactor was shutdown by safety system action. The 48V dc distribution panel was quickly re-energized by manually switching in a backup supply; some unit cooling was restored when a backup supply from the common service water system was manually valved-in; and the affected class IV buses were re- energized following field inspections.

There were no unusual releases to the environment and no personnel overexposures as a result of this event. The utility investigation into the root cause(s) is still in progress, but the utility has already begun implementing changes to correct some of the deficiencies identified as a result of this event. These include a review of the ground fault tracing practices, an operating manual review, and a review of the requirement to periodically test the common service water tie-valves to confirm their availability.

Report # 1493G0 PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1993 - A STATUS REPORT (NUREG/CR-4674, ORNL/NOAC-232, Vol. 19 and Vol. 20)

Under NRC's Accident Sequence Precursor (ASP) Program, operational events that involve portions of postulated core damage sequences are identified and evaluated using event tree models and probabilistic risk assessment (PRA) techniques. The event trees represent plant equipment and human actions that could affect, or be used to mitigate, the event being evaluated. ASP allows quantitative estimates of the significance of the event to be made in terms of a conditional core damage probability. This IRS report discusses the ASP analyses that were performed on events which occurred in 1993.

Sixteen events were identified as precursors having conditional probabilities of subsequent core damage of 1E-6 or higher. Twelve were associated with pressurized water reactors (PWRs) and four with boiling water reactors (BWRs). A number involved problems with electrical systems, with loss-of-offsite power events occurring at four plants. The precursors were evenly divided between those involving the unavailability of safety equipment and those involving initiating events, with the distribution of precursors by conditional core damage probability in the two categories being approximately the same. Seven of the eight precursors associated with unavailabilities occurred at PWRs, whereas five precursors associated with initiating events occurred at PWRs and three at BWRs. Twelve of the sixteen precursors occurred at multi-unit sites, but only one affected both units at a dual-unit site.

The following four events had conditional core damage probabilities greater than 1E-4 and therefore were considered important: (1 and 2) potential unavailability of essential service water at xxxx Units 1 and 2 discovered on February 25, 1993; (3) scram and loss-of-offsite power at xxxx Unit 1 on September 14, 1993; and (4) clogged suppression pool strainers and service water flood at xxxx on March 26, 1993.

Report # 1553G0 MALFUNCTION IN MAIN GENERATOR VOLTAGE REGULATOR CAUSING OVERVOLTAGE AT SAFETY- RELATED ELECTRICAL EQUIPMENT (NRC Information Notice 94-77)

This IRS report discusses how a malfunction in the main generator voltage regulator could increase generator output voltage, which could cause an overvoltage condition at the vital buses powering safety-related electrical equipment. On March 1, 1993, a steam extraction line to a feedwater heater ruptured at the xxxx Unit 2 Nuclear Plant. Heat and moisture from the steam rupture caused the main generator voltage regulator to malfunction, increasing the excitation to the main generator (overexcitation). The overexcitation caused the main generator voltage to rise about 19 percent above the normal output voltage. Operators tripped the main generator manually because they could not control the output voltage, which caused turbine and reactor trips. Although the overvoltage transient caused the voltage at the vital buses to rise, the licensee determined that, in this instance, the voltages were within the design limits of the equipment involved. A volts/hertz relay had been installed to protect the generator and transformers, but it was only used to initiate an alarm in the control room. On April 16, 1993, the xxxx Unit 2 Nuclear Plant automatically tripped while at 100-percent power in response to a main generator exciter field breaker trip because of overexcitation (overvoltage). The overexcitation was caused by a malfunction in the voltage regulator circuitry. At this plant, a volts/hertz relay initiates an alarm And also trips the main generator exciter field breaker. In addition, the transformers that supply power to the vital buses from the grid at xxxx are equipped with automatic load tap changers that maintain the proper voltage at the vital buses. Main generator protective relays, designed only to activate an alarm and not to trip the field breaker in response to overexcitation, may not protect the associated vital electrical equipment from voltages beyond the design limits of the equipment.

Report # 601200 INSTRUMENT AIR FAILURE AT TAPS

A decrease in instrument air pressure initiated the rod drive instrument air low pressure alarm for both units. Rods were not drifting in. Main LCV had closed from 60% to 40% and reactor level was going down. Reactor recirculation pump "B" was tripped manually. However, level could not be controlled, due to lack of adequate communication between LCV room and control room. Reactor level went down further and scrammed due to deenergisation of reactor low level relays. 1 1/2 instrument air line going to the cooling water intake structure area had ruptured due to the surrounding saline atmosphere causing severe corrosion. The leak was isolated by closing valves on the instrument air header loop system. Additional isolation valve has been installed in the instrument air line section to provide proper isolation. Additional phones were also installed to improve communications.

Report # 602300 REACTOR TRIP ON ROD CONTROL SYSTEM FAILURE

During normal power operation, control rod G-7 dropped and therefore the power range high neutron flux negative rate reactor signal was actuated. G-7 rod assembly is located approximately in the centre line of the reactor core. Reasons for the occurrence was the development of corrosion in the eye bolt of the vent plug in the central rod drive mechanism that enabled the reactor coolant leak causing a short circuit to the stationary gripper coil. As a corrective action, the 0-ring gasket and a blocking diode were replaced and moisture removed from the inside.

Report # 603900 EMERGENCY FEEDWATER SUPPLY ACTUATED

At 18.18 during a periodic operating test a feedwater pump was switched off and the check valve on the pressure side of the pump failed to close due to an erosion problem. The feedwater pump in reserve was automatically started, but soon all feedwater pumps were tripped by low pressure protection. The steam generator levels reached the "low-low" setpoint, and the reactor protection and ESF systems were actuated. The upper tube bundles in the steam generators were uncovered, and approx. 4 m3 of relatively cold water was fed into the steam generators by the emergency feedwater pumps, until the personnel succeeded in restarting the feedwater pumps and managed the situation. No signs of damage to the steam generators were found and the unit was started. During the annual shutdown the emergency feedwater nozzle areas of the steam generators were examined.

Report # 604700 SHUTDOWN OF THE REACTOR DUE TO OPENING AND NON-CLOSURE OF A PILOT-OPERATED RELIEF VALVE

During trial operation with electrical output 300MW the reactor was shut down by the first order emergency system (AZ-1). The emergency system was triggered by an alarm signal indicating a pressure drop below 15MPa in the primary circuit due to spurious action of the pilot-operated relief valve of the pressurizer, that stopped in the open position. There was a rapid drop in pressure in the primary circuit to the point of saturation. The main circulation pumps were stopped. The core emergency cooling system came into action. The reason for the false relief valve operation was a fault in the control circuits. The proper testing of safety valves operation was carried out using a switch on a different control panel.

Report # 605400 RCS OVERCOOLING RESULTING FROM A FAILURE TO CLOSE TURBINE BYPASS VALVES

On January 3rd, a faulty steam pressure sensor for turbine control indicated a higher value of steam pressure, which resulted in turbine bypass valves opening and steam pressure dropping to 2 MPa. Operators did not notice the alarm indication of the turbine bypass valve. This caused a rapid decrease of coolant temperature in RCS, and subsequently both turbines tripped and the reactor scrammed. Operators closed the main steam line block valves and the event was terminated. The unit was then started up following an investigation which had not revealed the cause of the incident. Two days later, the event recurred, but with milder consequences. During the event seven other failures also occurred. The report refers to four other incidents in Units 1 and 2 in which turbine bypass valve failures or cooldowns on RCS occurred. Reference is also made to related incidents in IRS 6030, 6031 and 6055. The two events indicate the significance of thermal shocks which may result from failure on the secondary side and the importance of identifying faults prior to starting up following reactor scrams. Corrective actions include better indications in control room, modifications on the steam lines, providing automatic interlocks and improving emergency procedures.

Report # 607600 STUCK OPEN INSTRUMENTED RELIEF VALVE IN MAPS UNIT NO. 1

MAPS Unit 1 was operating at approximately 10% full power on June 25, 1986. The turbine was spinning at rated speed and new generator transformer drying out was in progress. At 02.13 hours, the reactor tripped on high pressure in the Primary Heat Transport (PHT) system. After the reactor trip, the reactor pressure became normal, but increased again soon after. The increased pressure caused the three instrumented relief valves (IRVs) to open and release heavy water into the bleed condenser. One of the IRVs remained open for 548 secs. and then closed back automatically. The rupture disk of the storage tank blew, resulting in approximately seven tons of hot heavy water spilling into the boiler room. The containment isolation dampers closed automatically. The most probable reason for the pressure rise is the blocking of the bleed path either due to malfunction of bleed valves or bleed condenser inlet valve/bypass valve resulting in a pressure build-up due to continued in-feed through the circulating pump glands. Investigation could not establish the reason for the delayed closing of the relief valve. The following corrective actions were taken: - Reassessment of feed and bleed system; - Provision of recorders for bleed condenser pressure and bleed inlet and outlet temperature; - Provision of position indicator of all the valves; and - Review of component design and associated control logic of storage tank.

Report # 608600 POWERING DOWN OF NO. 2 REACTOR AT NOVOVORONEZH NPP ON 08.03.87 OWING TO A FAULT IN THE CONTROL SYSTEM DRIVE MECHANISM

While No. 2 reactor at the xxxx NPP was operating at a power of 345 MWe, a fault in the electrical connectors of the scram assembly drives caused the assembly to drop without warning into the core. In accordance with the operating regulations, reactor power was reduced to 250 MWe. The fault was caused by a decreased isolation between the pins of the connector to aging. There was no direct safety consequences. In the future all electrical connectors will be changed at the same time as the assembly drive mechanisms.

Report # 609500 DISRUPTION OF NORMAL OPERATING CONDITIONS AND EMERGENCY SHUT-DOWN OF THE FIRST UNIT OF THE VVER REACTOR OF THE xxxx NPP DUE TO FIRE

On 15th October 1982, a non-self-isolating short circuit in the terminal box of the electric motor of a process-water pump in the reactor building resulted in overheating and damage to the 6-kV power cable of the motor. Fire, which spread to the nearby cable lines of the unit control panel, led to disruptions in the operation of some of the control and regulation systems. Thanks to the fault-clearing action of the operating personnel and fire prevention service the fires were extinguished, and emergency shut-down of the reactor and cooling of the primary circuit were carried out. There was no disruption of core cooling. The radiation levels were within the permissible limits of radiation safety for normal operating conditions. The power generating unit remained shut down until completion of repairs to the damaged cable lines and revision of systems and equipment that were affected by the non-design-basis operating conditions during the emergency shut-down of the reactor and the cooling of the primary circuit. Additional safety measures were also introduced.

Report # 609600 PRESSURIZER SPRAY CONTROL VALVE FAILURE AND DEPRESSURIZATION OF REACTOR COOLANT SYSTEM

Regular daily dilution of boron in the reactor coolant system (RCS) necessitated the manual energisation of the pressuriser heaters in order to equalise boron concentration in the RCS and the pressuriser. Immediately after that manual operation, pressuriser spray control valve PCV 655B opened automatically due to pressure increase, and failed to re-close due to orifice plugging caused by dirt in the instrument air system. Fast RCS depressurisation led to immediate diagnosis of the event. Load reduction was initiated and was followed by an automatic reactor scram on low-low pressuriser pressure. Eight seconds after the scram, the high pressure emergency core cooling was actuated automatically. RCS subcooling was maintained. Safety injection was terminated fifteen minutes after its actuation, when the pressuriser went solid. Twenty minutes after the incident started, plant conditions were restored to the normal cold shutdown state. The root cause of the event was the installation of an incorrect filter regulator on the valve. Follow-up actions include increasing instrument air surveillance, replacing the filter regulator and improving the design and QA procedures related to that system.

Report # 611500 LOSS OF OFF-SITE POWER WITH NON-AVAILABILITY OF EMERGENCY POWER SUPPLY

Unit 2 was operating at a power of 960 MWe. Owing to condensation of moisture on the support-type insulators of the 24 kV line in the circuit serving the electrical power system, short-circuiting to earth of the power line insulation occurred. The anti-shorting protection came into operation, and the 750 kV air-break switches and the unit transformer were disconnected. The generator was switched off, and the turbine emergency stop valves closed. The reactor was stopped by actuation of protection system AZ-1. With the current supply cutoff the diesel generators started up, and the safety system mechanisms came into operation. Core cooling was effected by natural circulation. Residual heat removal from the reactor proceeded normally. The limits and conditions of safe operation were not violated. The radiation situation at the nuclear power station was normal. Owing to the absence of an emergency supply to meet internal load (withdrawal for repair), disconnection of the emergency power supply - the diesel generators - required 6 kV. The cause of the damage to power line was attributed to faults in design, manufacture and technical servicing. The cause of the withdrawal for repair of all the reserve power supply sections was a circuit design fault. As a result of this incident and of earlier failures and damage, substantial corrective actions were taken. The incident had no effect on the operation of the other station units.

Report # 611700 DAMAGE TO FAST ACTING GATE VALVE OF THE PRESSURIZER WATER INJECTION SYSTEM

With the unit operating at a power close to rated level, operation of the pressure control in the pressure compensator (PC) induced oscillations in the pressure in the primary circuit within the range 164 kg(f)/cm2 - 156 kg(f)/cm2. When the pressure stabilized at the latter figure, the signal: "pressure increase in containment more than 1.003 kg(f)/cm2" was triggered, and the containment ventilation system valves closed. The cause of the pressure drop in the primary circuit could not be established, and the operating staff proceeded to reduce power. At a power level of 740 MWe the reactor emergency protection system (AZ-1) came into action through the pressure reduction in the primary circuit, and the unit closed down. Investigation of the incident indicated that damage had occurred to one of the gate valves of the pressure compensator water injection unit. This was due to a design fault and also to errors in the way in which the technological system instrumentation was used. In violation of the technical requirements for gate valves of this type, the design made no provision for drainage of normal leakage from the inter-gland space. This resulted in release through the sealing, and to corrosion and destruction of the attachment unit. Following the investigation, measures were taken to eliminate the design and operating faults. After repairs of the pressure compensator assembly, the unit was returned to its normal operational state. The radiation situation in the containment remained within permissible limits and there was no radioactive exposure of personnel.

Report # 612100 LOSS OF POWER IN 6.6 KV EMERGENCY BARS

During normal plant operation, due to a failure in the hydraulic turbine excitation system, an overcurrent originated a loss of power in one of the two emergency bars. The BV bar remained without power during 7 min 53 sec due to the existing interlocking to protect the bar and the time necessary to investigate the causes of the failure. As a result of the performed inspections, it was determined that the initiating event was due to premature wear of the hydraulic generator brushes.

Report # 612400 INADVERTENT ACTUATION OF SAFETY SYSTEMS RESULTING IN SCRAM

In the course of a loss-of-voltage automatic transfer test at 20% power during commissioning, emergency 6kV buses disconnected inadvertently from auxiliary transformer and safeguard load sequencers as parts of safety systems actuated on the resulting false signal of loss of off-site power. The load sequencers closed the valves in the outlet lines of pressurized leakages from main coolant pumps sealing systems, as if the pumps had tripped. Due to the closing of these valves, 5 out of 6 pumps actually tripped, and this resulted in reactor scram. The event is a recurrent example of an inadvertent reactor scram caused by a spurious actuation of safety systems due to an incorrect setting of safety parameters.

Report # 613600 OPENING OF ALL FAST-ACTING STEAM DUMP VALVES AT A SPURIOUS SIGNAL AND FAILURE OF THE TURBINE CONTROL SYSTEM

The unit was operating at 960 MWe power with nominal parameters. During the replacement of faulty electronic block in the measuring and alarm circuits of secondary coolant, all four turbine bypass steam discharge valves (TBSDVs) suddenly opened releasing steam into the condensers. This resulted in the pressure decrease in the secondary. Turbine control system started to reduce turbine power. Operational personnel performed remote closure of all TBSDVs from automated process control system panels. Pressure increase in the secondary began. Turbine control system did not respond to the pressure increase and continued to reduce the turbine power. The pressure in the secondary circuit reached the setpoint of reactor emergency protection actuation (AZ-1) and the unit was shut down. After unit shutdown efforts were undertaken to investigate the causes of spurious signal for TBSDVs opening and the failure of the turbine control system. A defect in the new electronic block was found and a fault was made by the manufacturer during assembly of turbine control system switch. After the repair of the discovered defects, the unit was returned to service. Violations of safe operation limits and conditions did not occur. Personnel actions were correct. As a corrective measure, the discovered defects were repaired and changes were made to the procedure of acceptance testing of electronic blocks delivered to the plant.

Report # 614300 NON-CLOSURE OF MAIN SAFETY VALVE DURING REGULATORY TESTING AND ADJUSTMENT PRIOR TO START-UP

The reactor was in the start-up phase. Prestart testing after scheduled maintenance before power raising was underway. During testing and adjustment of the main safety valves (MSV) in the main steam lines of live steam, one of the valves did not close. After unsuccessful attempts to close the valve, the reactor was scrammed using the emergency AZ-5 button. This resulted in the uncontrolled cooling transient at the rate of 53 deg-C/hr. The cause of the failure to close was a faulty pilot valve (PSV) used for MSV control. However, on 1 Aug. 1987, during MSV regular testing and adjustment before unit start-up, the same valve failed to close again. The reactor was again scrammed which was followed by the recurrence of the cooling transient at the rate of 33 deg-C/hr. Detailed inspection of the valve revealed differences between actual and design dimensions of seat surfaces of the valve components (cover, stem, sleeve) which caused the stem sticking after valve heating by the steam. Violations of design limits and safe operation conditions did not occur. Radioactive discharges were not observed because radioactive steam was drained to the pressure suppression pool. After repair of valve defects and subsequent satisfactory operability tests, unit power raising was continued. As a corrective action all the remaining pilot valves were inspected.

Report # 614700 FAILURE OF THE PRESSURIZER INJECTION VALVE

The unit was operating at nominal power. During the leveling of boric acid solution concentration in the primary circuit, the pressurizer injection valve drive supply breaker failed causing pressure decrease in the primary circuit. Personnel made a successful attempt to close the pressurizer injection valve remotely. Due to a pressure drop in the primary circuit, reactor emergency protection actuated. The causes of the event were: A) failure of pressurizer injection valve drive supply breaker during the levelling of boric acid solution concentration in the primary circuit; b) interlocking of valve closure in the pressurizer injection line during pressure decrease in the primary circuit failed to actuate (due to the absence of information on pressurizer injection valve position). This resulted in reactor emergency protection actuation. Violations of safe operation limits and conditions did not occur. Measures have been taken to improve the interlocking operation algorithm for closing the valve in the pressurizer injection line. The modification of the instrument portion of the generation scheme of the protection signal "Primary Circuit Temperature Deviation from Saturation Temperature Less Than 10 C" has been planned.

Report # 614800 REACTOR SCRAM DUE TO LEAK IN THE PRESSURE INSTRUMENTATION LINE OF THE PRESSURIZER LEVEL TRANSMITTER

While the unit was operating at nearly rated power, the emergency protection system actuated on pressurizer 10-10 level, resulting in the unit shutdown. The analysis of the causes of the emergency protection system actuation revealed that this resulted from the formation of spurious pressurizer low-low level signals. Spurious signals appeared due to a hole from under the coupling nut in the "negative" pulse line joint connected with one of the three pressurizer level transducers. Operator errors which contributed to the spurious actuation of the reactor emergency protection system are as follows: Improper repair of the hole in the pulse line resulting in the repeated leakage and untimely elimination of the repeated leakage. Moreover, the design principle of connecting all three level transducers to a single pressurizer bleed-off line does not rule out the possibility of generating spurious signals resulting from the damage of the common pulse line. However, expediency of changing this principle should still be considered. As a result of the performed event cause analysis corrective actions to provide proper equipment and systems inspection were taken. Violations of limits and safe conditions of operation were not observed during the event. Radiological situation in the unit rooms was within the permissible limits. Following analysis of the event causes and corrective actions, the unit rated power was restored.

Report # 616300 SPURIOUS OPENING OF PRESSURIZER RELIEF VALVE

During normal operation at nominal power, a spurious signal caused the actuation of the pressurizer pilot valve (PV) and the opening of the pressurizer safety valve (SV). Pressure in the primary circuit started to decrease. Emergency protection of the second kind (AZ2) actuated, followed by the actuation of emergency protection of the first kind (AZ1) and the reactor tripped. As the signal for pressurizer PV opening was not "cancelled," the pressurizer SV remained open and pressure continued to decrease. At 85 KGF/CM2 reactor emergency core cooling system actuated. Emergency diesel generators started and the pumps for emergency injection of boric acid were activated. On the signal of ECCS actuation primary circuit isolation valves closed and all main coolant pumps tripped. At 58 KGF/CM2 ECCS hydrovolumes actuated. Personnel deenergized the PV feed and the pressurizer SV closed. After stabilization of the primary circuit parameters, the safety systems (SSS) were turned off. During operation of the pressurizer SV the permissible rate of primary circuit pressure decrease was exceeded. Spontaneous opening of pressurizer PV followed by the opening of pressurizer SV resulted from a combination of two failures in the pressurizer PV control circuit. The root causes of the event were deficiencies in the conduct of design modifications of the pressurizer PV control circuit and deficiencies in cable sealing. After elimination of defects, replacement of control cable in the pressurizer PV control circuit, testing of all PV's control circuits and primary circuit hydrotests with equipment inspection, the unit was connected to the grid.

Report # 616400 UNPLANNED SHUTDOWN OF xxxx UNIT 2 ON SPURIOUS SIGNALS DUE TO CABLE FIRES

The unit was operating at 1250 MWe nominal power. A control cable fire occurred in one of the cable rooms of the reactor building which was automatically extinguished by the fire-fighting system provided by the design. The fire damaged about 650 control and monitoring cables. This led to a reactor shutdown by emergency protection due to an MCP trip on spurious signals and to partial loss of parameter monitoring and unit equipment control capability. The unit was cooled down at the rate specified by the operation procedure. Due to cable damage, safe unit operation conditions were breached as there was a temporary loss of capability to monitor the status and parameters of several safety-significant systems of the plant. Radioactivity releases and personnel injuries did not occur. Personnel actions to cope with the event were correct. After cooldown, the unit was taken out of service for repair and recovery operations. Possible causes of the fire were: A) mechanical damage to the control cable in the location of the bend due to deviations from standard requirements for bend radius during assembly of cable ducts and/or b) continuous cable heat-up due to overcurrent flow or external short-circuiting and failure of protective device. Based on the results of investigation, appropriate corrective actions have been taken and planned.

Report # 617400 ACTUATION OF THE FIRE FIGHTING SYSTEM AND ONE CHANNEL OF THE SAFETY SYSTEM DUE TO SPURIOUS SIGNALS

The unit was operating at 995 MWe power. A spurious signal caused the actuation of automatic fire extinguishing facility which supplied water into the cable duct of one of the three control channels of the safety system (SS). A defect in the insulation of the power cables of the SS control channel led to the failure of the power supply units feeding this SS channel. The mechanisms of the second SS channel actuated. The pneumatic isolation valves of the primary circuit closed. Protection on oil pressure decrease in the MCP oil system tripped MCPs 1 and 3. Accelerated unit power reduction occurred. This was followed by an MCP 2 trip for the same reason as MCPs 1 and 3. Reactor emergency protection actuated, the turbine generator tripped. The unit parameters were subsequently stabilized. Violations of safe operation limits and conditions did not occur. There were no radioactivity releases or personnel exposure. The root cause of the event is deficiencies in the start-up commissioning activities which did not include complete high voltage testing of the cables. As a result, latent defects of SS power cables insulation were not discovered in time. After detection and elimination of the defects, normal unit operation was restored. The following corrective actions were taken as a result of event analysis: The manufacturer will develop measures to improve the reliability of fire detectors. A system of drains from the ventilation facility rooms will be introduced to prevent process equipment leaks into the cable ducts and rooms.

Report # 620400 LOSS OF POWER IN THE ACTUATION CIRCUITS FOR POWER SUPPLY TO THE REACTOR CONTROL AND PROTECTION SYSTEM

The unit was operating at nominal parameters and with normal electric circuit. Due to voltage oscillations in the third channel of the auxiliary power system for consumers of Category 1 reliability, a decision was taken to repair the inverted converter of the continuous power supply set. As a result of this activity, Category 1 reliability consumers (components) were transferred to standby power supply source. This transfer was accompanied by magnetization current inrushes in the consumer circuits causing deenergizing of a number of panels including the reactor protection system (RPS) power supply control circuits. The AZ1 emergency protection system actuated and the unit was shut down. Breach of safe operation margins and conditions did not occur. Six hours 38 mins. later after discovering the failure causes, the power supply circuit for Category 1 consumers was restored and the unit was connected to the grid. The root cause of the event was design deficiencies in the system of power supply for Category 1 consumers. As a result of event analysis, additional tests have been planned for the auxiliary power system for Category 1 consumers including the improvement of the electrical circuit of RPS power supply control panels.

Report # 6255G0 GENERIC PROBLEM OF LOAD REJECTIONS

During operation, eight events with complex load rejections after unit disconnection from the grid with reactor power reductions and fast automatic transfers of power supply to startup transformers occurred. During each event, multiple (both dependent and independent) occurrences took place. Half of the events resulted in scrams, the others terminated at low reactor power. The safety significance of this generic problem consists in a possibility of a loss of off-site power in case of an actuation of safety systems. The problem was rated as level 1 in the INES scale due to the frequent recurrence of complicated load rejections at the xxxx units.

Report # 6256G0 DISCONNECTION OF ALL xxxx UNITS FROM GRID.

A short circuit occurred in an external 400 kV switching station due to an error by an electrician. It resulted in a disconnection of both 400 kV and 110 kV lines to all four xxxx units. The power supply in both systems was restored within 13 min. Unit 1 failed to cope with the load rejection due to false actuations of distant protections at its generators, and it resulted in a loss of off-site power with fast reactor scram HO-I from the loss of all MCPs. The HO-I scram from tripping the last operating turbine failed due to a failure of both stop valves at one TG to close completely. There was a large number of occurrences at Unit 1. Unit 2 coped with the load rejection automatically and reduced its power down to the level of self consumption of 25%, but with a number of occurrences. Unit 3, which had been shut down for refueling prior to the event, lost power supply from its start-up transformer and one out of its two available DGs failed to start due to a failure in its automatic excitation but started successfully within 1 min following its manual excitation. Unit 4 coped with the load rejection automatically, and reduced its power down to a level of 20%, but with a number of occurrences. The significance of the event for safety consists in: - The occurrence of a common-mode failure with a disconnection of all four units from grid due to a single fault in the switching station; - the failure of one unit to cope with the load rejection leading to scram; - the failure to scram from tripping the last operating TG; - the loss of off-site power; - the failure of one available DG at a unit shut down for refueling to start automatically; - 45 occurrences at all units during the event.

Report # 6274G0 UNPLANNED SHUTDOWNS OF VVER-1000 PLANTS DUE TO DESIGN DEFICIENCIES IN SERVICE WATER SYSTEMS FOR ESSENTIAL EQUIPMENT

In 1990 and 1991 a number of events occurred at VVER-1000 plants (V-320 and V-338 model) causing unit scram due to reactor building essential service water system (ESWS) design deficiencies (protection and interlock panels flooded by ESWS emergency storage tank water). This caused 'spurious' actuation of protection and interlock sensors of secondary circuit equipment which led to automatic reactor scram. The causes of the emergency service water storage tank overfilling were personnel wrong action and/or check valve design deficiency. Breach of limiting conditions of operation and radioactivity releases did not occur. To prevent the recurrence of similar events corrective actions have been taken to: Improve quality of personnel training; increase frequency of check valve maintenance; install protective covers above the protection 7 interlocking system panels, etc. According to the International Nuclear Event Scale (INES) these events are rated as level 1.

Report # 627700 ACTIVATION OF THE EMERGENCY REACTOR PROTECTION SYSTEM DUE TO SPURIOUS SIGNAL DURING DEENERGIZATION OF THE SKALA CENTRALIZED MONITORING SYSTEM

During unit normal operation at nominal power, due to operating personnel errors a number of instruments of "Skala" centralized unit parameter monitoring system were deenergized. Because of a circuit deficiency of the "decreased feedwater flow" protection a spurious signal was generated causing a reactor scram. The spurious signal of the loss of feedwater flow at nominal power occurred as a result of delayed return to service of different parameter monitoring instruments during interruption and subsequent restoration of the power supply. Further personnel actions were, as a whole, correct and no deviations from procedures were observed. The root causes of the event were deficiencies on the organization of operation, shortcomings in personnel training, and a protection system circuit design deficiency. Breach of unit limiting conditions of operation occurred since during the period of deenergization of some of the Skala system components (3 min.) One out of the four trains of technological protections was rendered inoperable. Based on the results of the event analysis, actions have been taken to improve the reactor (emergency) protection systems circuit and to eliminate the observed operation deficiencies. According to the International Nuclear Event Scale (INES) this event is rated as level 1.

Report # 630400 INADVERTENT REACTOR COOLANT SYSTEM DEPRESSURIZATION DUE TO PRESSURIZER SPRAY VALVE FAILURE

The plant was restarting after turbine repair work. When reactor power was increased to 1.5% in order to supply main steam to the secondary side, the pressurizer pressure decreased abnormally. Immediately the operator closed the Main Steam Isolation Valve (MSIV) and turned on the heaters of the pressurizer to recover the primary pressure. However the pressurizer pressure continued to decrease, and consequently safety injection was actuated due to low pressurizer pressure. The plant was brought to a controlled state with actuation of safety injection. Subsequent investigations revealed that pressurizer spray was inadvertently supplied due to the failure of the link bar of the valve actuator unit. The pressurizer spray was interrupted by valve closure on a containment isolation signal.

Report # 633100 BATTERY MALFUNCTION DETECTED DURING TESTING

The unit was operating normally at 50% of nominal power. A turbine feedwater pump (2TPN-1) was undergoing maintenance. Personnel had started scheduled testing of the emergency power supply for train 2 of the safety systems.

Due to battery malfunction the voltage in the DC board dropped to the tripping setpoint of two inverters, and until the power was restored using diesel generator Category 1 reliability components, remained deenergized. Spurious actuation of protections "SG level drop to -500 mm below nominal" and "temperature difference Ts <10 C" occurred in all the four loops of the reactor facility. The reactor coolant pumps (RCPs 1,2,3, and 4) tripped and a reactor scram followed.

After replacement of faulty elements in the battery unit operation was resumed.

Personnel should improve the quality of battery maintenance. The designer should upgrade the design of the SS emergency power supply to eliminate deficiencies revealed while in operation.

According to the International Nuclear Event Scale (INES) this event is rated as Level 0.

Report # 6339G2 TOTAL LOSS OF POWER AT KOLA NPP UNITS CAUSED BY GRID DISTURBAN- CES DUE TO TORNADO

Prior to the event xxxx NPP was in normal operation at the following power levels: Unit 1 - 370 MWe, Unit 2 - 290 Mwe, Unit 3 - 350 MWe, Unit 4 - 365 Mwe. Safety systems were in hot standby.

Due to the hurricane wind the "xxxx" grid system collapsed, and the 330 kV, 154 kV, and 110 kV high voltage transmission lines were damaged. Spiked voltage oscillations in the NPP unit auxiliary power line resulted in trips of the turbine generators and other main equipment and reactor scrams.

An attempt to supply power to Unit 1 and 2 (WWER-440, V-230 designs) equipment by emergency connections - from diesel generators (DGs) was unsuccessful due to DG failure for the following reasons: Deficiencies in DG control design configuration and deficiencies of work planning and organisation by NPP management as regards timely changes of DG control design configuration and ensuring emergency power supply to essential equipment. Total blackout of these units was accompanied by breach of safe operation limits and conditions.

Safety systems at Units 3 and 4 (WWER-440, V-213 designs) are configured as three channels with independent power supplies, service water supplies, compressed air supplies, etc. For this reason total blackout conditions at Units 3 and 4 passed without serious criticism.

According to the International Nuclear Event Scale (INES) the events at Units 1 and 2 are rated as Level 3 and the events at Units 3 and 4 are rated as Level 1.

Report # 634400 DEFICIENCIES IN THE ORGANIZATION OF STAND-BY DIESEL AND SAFETY SYSTEM BATTERY OPERATION

On 92.09.03 Unit 1 was operating at nominal power. During routine testing of safety system (SS) Train 3, DG-3 failed to start on a signal of 6 kV essential bus deenergization. The DG-3 failure occurred due to degraded DG battery voltage. At the time of the DG-3 battery failure the service lives of the batteries of DGs 1 and 2 had expired.

On 91.12.27 Unit 2 was in normal operation at 50% nominal power. During routine testing of SS train 2, a reduced battery voltage occurred, leading to loss of power to equipment automatic control features and protection and interlocking sensors. After connecting DG-2 to the 6 kV essential bus, power to essential buses was restored and equipment automatic control features were energized. A reactor scram followed.

The root causes of the events were: lack of battery lifetime monitoring capability, failure to take timely action to replace batteries with expired service life, battery maintenance deficiencies.

The events were dealt with by replacement of batteries. Operational documents were amended as regards the scope and frequency of battery status monitoring efforts. Violation of unit safe operation conditions did not occur.

According to the International Nuclear Event Scale (INES) these events are rated as level 0.

Report # 635000 DE-ENERGIZATION OF DC SWITCHBOARD DUE TO DAMAGE TO A REVERSIBLE MOTOR GENERATOR AND BATTERY TRIPPING

The unit was at a thermal power of 3% of rated (Nnom) power with TG-2 at coast-down and TG-1 under maintenance. A short circuit occurred in the armature winding of the reversible motor-generator (RMG-1) connected to the unit DC panel (UDCP-1). The automatic breaker which disconnects UDCP-1 from RMG-1 failed to open, and the DC grid continued to receive voltage which significantly differed from the nominal. The resulting current oscillations led to the opening of the automatic breaker which supplies power to RPS emergency protection panels from UDCP-1. The automatic breaker of the same panels fed from the common unit DC panel opened spuriously. Because of loss of voltage on the RPS panel, AZ1 protection actuated and the reactor was scrammed. A signal for diesel generator (DG) start was generated. However, the diesels did not start since (because of the UDCP-1 deenergization) the working DC supply to the control circuits of some safety system mechanisms having no standby power supply from other DC panels was lost, including the DGs. This resulted in a failure to start two service DGs and one standby DG, loss of remote control function of the safety injection pumps, the emergency feedwater pumps and the SG safety valves. Twenty minutes later, power to UDCP-1 was supplied through the mutual redundancy grid from the common unit DC panel. Reactor parameters remained under control.

In this event breach of safe operation conditions occurred, since after AZ protection actuation, loss of some safety system mechanism control functions existed for 20 minutes. Personnel exposures or radioactive product releases did not occur. The root causes of the event were design deficiencies in the configuration of the DC supply and deficiencies in the organization of equipment operation and maintenance. To prevent similar events, modifications to the DC grid have been envisaged.

According to INES this event is rated at Level 2.

Report # 635400 REDUCTION IN FEEDWATER FLOW RATE OWING TO OPERATOR ERROR AND DEFICIENCIES IN PROCEDURE

While at 700 MWe power one control rod dropped due to rupture of a control rod drive (CRD) ribbon. Because of insufficient insertion depth, the Local Automatic Control (LAC) rods failed to compensate the inserted negative reactivity, which resulted in a decrease of power in one reactor half (side) by 250 MWt and a steam drum level reduction.

The operator erroneously interpreted this as a control feature fault and initiated manual control of the feedwater governor, which led to a feedwater flow-rate reduction below 75% of the current level, causing the actuation of AZ-1, AZ-T2 and AZ-5 emergency protections and a reactor scram.

No breach of safe operation limits or conditions occurred. Neither radioactivity releases nor personnel overexposures were observed.

The root causes of the event were the poor quality of the equipment maintenance and monitoring practices, deficiencies in operating personnel training, and deficiencies in the operational documentation.

To prevent any recurrence of similar events, actions have been taken to improve personnel training, enhance the quality of RPS equipment, maintenance, and monitoring, and upgrade the operational documentation.

According to the International Nuclear Event Scale (INES) this event is rated as Level 0.

Report # 635900 UNAUTHORIZED ACTUATION OF A PRESSURIZER PILOT-OPERATED RELIEF VALVE

The unit was in normal operation at 325 MWe power. With unchanged primary circuit parameters, a brief opening of a pressurizer pilot-operated relief valve (PORV) occurred. Subsequently, until the PORV electric circuit was disconnected, several PORV opening/closing actions followed. The unit was scrammed by the AZ-1 protection on a "small leak" signal. All safety systems functioned as designed. After disassembly of the PORV's electric control circuits, pressure oscillations in the primary circuit stopped. The primary circuit parameters were stabilised.

The unauthorized opening of the pressurizer PORV occurred due to a short circuit between cable conductors in the cable penetration because of degraded isolation resistance between the conductors. The degraded resistance between the control cable conductors was caused by loss of leak-tightness and moisture ingress into the penetration due to a penetration seal design deficiency.

The root cause was a design deficiency: lack of physical segregation between cable conductors of the cable from the ECM to the PORV control logic.

No breach of unit safe operation limits or conditions occurred. Analysis of the primary circuit parameter changes showed that the rate of reactor cooldown was in correspondence with the cooldown rate at scram actuation.

According to the International Nuclear Event Scale (INES) this event is rated as Level 1.

Report # 636000 REACTOR SCRAM ON HI-HI STEAM DRUM LEVEL DUE TO DEENERGIZATION OF THE 0.4 KV BUS

While at nominal power the 6.0/0.4 kV essential power transformer tripped, causing the 81NNB 0.4 kV bus to deenergize. Because of the absence of standby power, all the loads connected to this bus were deenergized, including the oil pump of the turbine and the bus which feeds the valves of the facilities which supply water to the steam drums. Due to this loss of power to the oil pump, combined with a failure of the standby pump to start, a level decrease in the Pressure Oil Tank (POT) occurred, plus a TG-8 trip by the POT lo-lo level protection. Reactor power was reduced to 1600 MWt by the protection. During the transient steam drum levels started to increase because the P1, 2-1312 control valves were deenergized and could not be remotely controlled. As a result the reactor was scrammed by the AZ-5 protection on hi-hi steam drum level.

No breach of unit safe operation limits or conditions occurred. The root causes of the event were design deficiencies in the circuit that feeds power to the feedwater governor drives and to the oil tank level control valves.

To preclude the recurrence of similar events, corrective actions have been planned to modify electric circuit configurations and replace oil tank level indicators.

According to the International Nuclear Event Scale (INES) this event is rated as Level 0.

Report # 636200 REACTOR COOLANT PUMP TRIPPING AT REDUCED SEALING WATER SUPPLY DUE TO FILTER CLOGGING

The unit was in the power raising phase after a Scheduled Preventive Maintenance (SPM). Turbine generator No. 4 (TG-4) was operating at 320 MWe power and TG-3 was being prepared for restart. All Reactor Coolant Pumps (RCPs) were in service. The water to the suction of the operating RCP Hydroseal Pumps (HPs-3,4) was being delivered from the HP storage tank. During TG-4 load increase an RCP sealing water flow-rate decrease occurred. Standby HPs 1 and 2 started, but this did not lead to any increase in the RCP sealing water flow-rate, thus causing the actuation of the Emergency Gas System (EGS), a trip of all RCPs, followed by a reactor scram by the AZ-5 protection system. No breach of unit safe operation limits or conditions occurred.

The cause of the reduced sealing water flow-rate was clogging of the grids of mechanical filters (MFs) installed at the suction of the HPs. The deposits which clogged the filters were ion-exchange resins (cationite) containing iron oxide, oil and silt/sludge impurities.

To prevent the recurrence of similar events, design changes have been made to the sealing water circuit and enhanced monitoring was introduced while performing "hydro-unloading" and ion-exchange resin reprocessing.

According to the International Nuclear Event Scale (INES) this event is rated as Level 0.

Report # 640300 ACTUATION OF THE REACTOR POWER LIMITATION SYSTEM AND CONSEQUENT REACTOR MANUAL SCRAM

Signaling the reactor limitation system actuation and control rods insertion has been registered. The insertion went on after switching to the manual control mode. A manual reactor scram was then performed. The failure in the reactor limitation system was a result of voting redundancy degradation caused by the misconnection in voting logic circuitry, which was not detected during periodic testing due to imperfect testing procedures. Then component (fuse) failure in one of measuring channels caused spurious actuation of the reactor control and limitation system.

Report # 640400 MANUAL REACTOR TRIP AS A RESULT OF FLOODING OF REACTOR CONTROL AND PROTECTION SYSTEM ROOM

In the course of current maintenance of HVAC equipment, the repairman let water begin to fill the tank and started activity on the other part of equipment. Water overflowed on the floor, which had been damaged and not repaired before, and leaked through the floor to the room where reactor control and protection system cabinets were located. Water affected low frequency converter and control rod position logic circuitry. As a result control rods accidentally dropped into the reactor core. The reactor was immediately tripped.

Report # 641200 LOSS OF UNIT AUXILIARY POWER AND LOSS OF POWER TO ESSENTIAL LOADS (CATEGORY 1 RELIABILITY)

The unit was in the regime of normal operation at 90% Nnom. A short circuit occurred during the destruction of phase "B" duct bus of AT-1-750 auto-connected transformer, and a protection of the above transformer disconnected the high voltage lines VL-330 "xxxx-1", VL-750 "xxxx" and the RTSN-2 Standby Auxiliary Power Transformer. A protection tripped generator G-1, and unit power reduction started. The differential protection tripped the VL-330 "xxxx-2" line, followed by total loss of power to the 330 kV open switchgear (OS-330), including RTSN 1 and 2 transformers. The normal leads of the 6 kV normal buses tripped and the standby ones were activated. Since the RTSN, 1 and 2 transformers were de-energized, the voltage in the 6 kV buses dropped to 0.25 Unom in approx. 1 s. This was accompanied by loss of power to all of the 0.4 kV and -220 V buses. During the transient the accumulator battery of the common DC board experienced a voltage drop, a fault occurred in the operation of inverters of the common unit uninterruptable power source, power was lost to a number of essential loads, and part of mimics and annunciator windows in the control room were disabled. As a result of de-energization of all RCPs, AZ-1 reactor emergency protection (scram) actuated. Due to the loss of auxiliary power, DGs 1, 2, and 3 started, and the load sequencing programs for SSs trains 1, 2, and 3 were initiated. The voltage on OS-330 kV and RTSN-1 transformer was restored in approx. 4 s, followed by the restoration of auxiliary power according to the normal configuration. No breach of safe operation limits and conditions occurred in the course of event. The causes of the event and corrective actions are described in the corresponding sections of this report. According to the International Nuclear Event Scale (INES) this event is rated as Level 0.

Report # 702500 ONSITE ANALYSIS OF THE HUMAN FACTORS OF AN EVENT AT DRESDEN UNIT 2 ON AUGUST 2, 1990 (SPURIOUS SAFETY RELIEF VALVE OPENING)

August 2, 1990, xxxx Unit 2 was at 80 percent power when a safety relief valve (SRV) opened spuriously and remained open. The control room crew successfully executed a manual reactor scram and plant cooldown, but with an excessive plant cooldown rate. The blowdown from the open SRV to the torus caused an initially rapid rate of temperature rise of the torus (0.72 deg C [1.3 deg F] per minute). The shift engineer, in command of the Unit 2 control room crew, followed a functional response mode of operation to this symptom and opened both turbine bypass valves (TBVS) for approximately two minutes following a successful scram of the reactor. This reduced the total heat input to the torus, but contributed to a 52.2 deg C (126 deg F) plant cooldown in one hour, which was in excess of the 37.8 deg C (100 deg F) per hour normal cooldown limit. NRC's analysis of this event demonstrated that plant cooldown without opening the TBVs would not have caused the torus temperature to approach its heat capacity temperature limit. The TBVs were closed at approximately 4,137 kPa (600 psig). Plant cooldown and decay heat removal, thereafter, was affected primarily by SRV blowdown to the torus, although all auxiliary steam loads were not secured until later in the event. Although spurious opening of an SRV is an anticipated event for a boiling water reactor, there was no event-specific guidance for plant cooldown in the plant procedures or training material. This is a report of the visit by a human factors team to analyze the control room crew operations during the event.

Report # 703200 REACTOR SCRAM DUE TO STEAM GENERATOR LEVEL SIGNAL DROP FOLLOWING NaOH INGRESS INTO TURBINE CONDENSATE LINE

During a filter regeneration in the steam generator blow-down purification station (SGBP) at full power, the control rod banks, No. 6 and 5 were inserted into the core by an actuation of the HO-3 slow scram from the signals of full-range level drops by -200 mm in at least two SGs. Increased turbine controller negative corrections (-), from main steam pressure drop, actuated and helped to reduce the outputs of both turbine generators (TGs). Two minutes after the HO-3, all remaining control rod banks fell into the core one after another from the HO-2 medium scram signal of reactor coolant system (RCS) pressure drop below 11.3 MPa. Operators then shut the reactor down by tripping both TGs manually.

The event was caused by a ingress of 800 - 1000 liters of NaOH solution via the condensate storage tank (CST) into the expansion tank of operational condensates (OPC) in the TG11 condensate system due to a valve left open in the course of SGBP filter regeneration because of a trainee's error. By exceeding the pH value of feed water for 7 minutes, a limiting condition was violated. Due to high concentration of NaOH in the feed water, the water in three SGs formed foam and level sensors indicated drops with the resulting HO-3 scram.