Systematic Assessment of U.S. Nuclear Power Plant Operating Experience Using the Sequence Coding and Search System

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Introduction

This white paper illustrates the use of the Sequence Coding and Search System (SCSS)[2] database for licensee event reports (LERs) to systematically examine the operating experience of domestic nuclear power plants for recurring events or other potentially adverse conditions or trends. The genesis of the white paper is the observations and recommendations of the NRC task force formed in response to the extensive degradation of the pressure boundary material of the Davis-Besse Nuclear Power Station reactor pressure vessel head[3], reflection on similar observations and recommendations of the Kemeny Commission Report on the Three Mile Island 1 accident in 1979[4], and observations made by Commissioner Merrifield at the March 2001 NRC Regulatory Information Conference.[5]

The nuclear industry has made verifiable progress in nuclear safety over the last decade. Plant availabilities are at an all-time high. The number of reactor trips has notably declined. The NRC has been responsive to industry calls for regulatory burden reductions supported by increased use of risk assessment tools and practices. A new Reactor Oversight Program (ROP) created with extensive industry and public input is in place. Nuclear plant economics are more attractive than ever before. Plans for additional nuclear plant orders are on the horizon. Commissioner Merrifield noted, "*I can say with confidence that the NRC and the industry have made substantial organizational changes to address lessons learned from TMI about the need for a questioning attitude*."[5]

What is the problem?

At about the time of the aforementioned conference, however, the severe nature of the Davis-Besse reactor head corrosion problem became known. In response to this problem, the NRC formed a task force to independently review the NRC's regulatory processes related to vessel head integrity and recommend areas for improvement pertinent to the industry and to the NRC. Subsequently, the Davis-Besse Lessons-Learned Report concluded that "*the event was not prevented because: (1) the NRC, DBNPS [Davis Besse Nuclear Power Station], and the nuclear industry failed to adequately review, assess, and followup on relevant operating experience…and (3) the NRC failed to integrate known or available information into its assessments of DBNPS's safety performance*." A number of the issues related to the review and the lessons learned from plant operational experience are hauntingly reminiscent of recommendations provided in the Kemeny Commission report over 20 years ago. According the Kemeny report (and as noted by Commissioner Merrifield), an attitude of complacency regarding plant safety arose following the several hundred years of plant operational experience with no member of the public being hurt. But, the investigation of the TMI accident revealed deficiencies with the "system" that manufactured, operated,

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[1] Managed by UT-Battelle, LLC for the U.S. Department of Energy under contract No. DE-AC05-00OR22725.
[2] *Sequence Coding and Search System for Licensee Event Reports*, NUREG/CR-3905, Vols 1-4, Oak Ridge National Laboratory, 1985.
[3] *Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head Lessons-Learned Report*, US Nuclear Regulatory Commission, September 30, 2002.
[4] Report of the President's Commission on the Accident at Three Mile Island, John G. Kemeny, Chairman, 1979.
[5] History's Lessons for our Future Challenges, Jeffrey S. Merrifield, Commissioner, U. S. Nuclear Regulatory Commission, 14th Annual Regulatory Information Conference, March 6, 2002.

and regulated nuclear power plants, including procedure deficiencies, training inadequacies, and failing to learn from previous events. As noted in the Kemeny Commission overview:

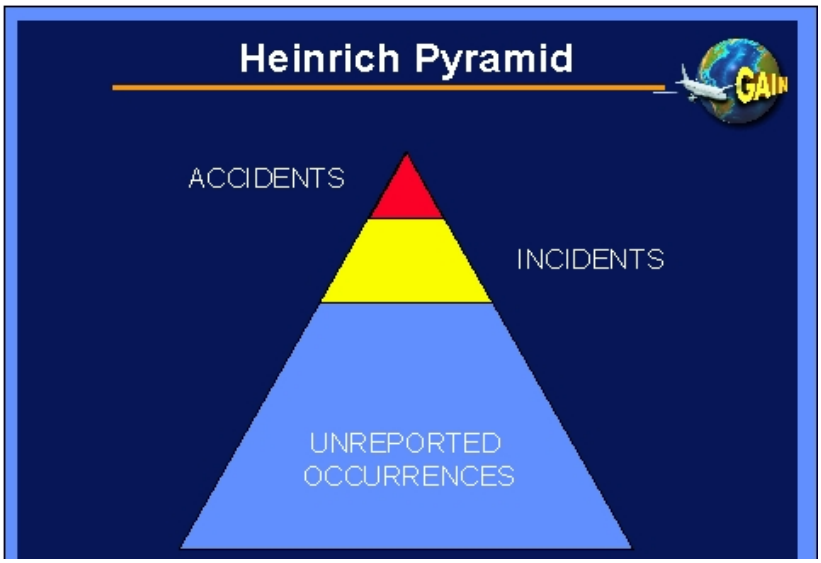
> *"We find that there is a lack of "closure" in the system – that is important safety issues are frequently raised and may be studied to some degree of depth, but are not carried through to resolution; and the lessons learned from these studies do not reach those individuals and agencies that most need to know about them."*[4]

The Davis-Besse Lessons-Learned Report also noted NRC organizational changes that adversely impact operating experience review programs. Until 1999, for example, NRC's Office for Analysis and Evaluation of Operational Data (AEOD), formed following the TMI accident, performed longer term operating experience reviews and assessments of operating failure trends and lessons learned. As noted in the Davis-Besse report, operating experience reviews currently performed do not include these issues. Numerous other assessments and reports based on operating experience were also eliminated or assigned to NRC's Office of Nuclear Regulatory Research (RES), which was to provide focused analyses of operational data and not expend NRC Staff resources on events that are not risk significant. The report also notes (on p. 40) that while RES performs these studies or evaluations, they are generally not initiated without a request from NRC's Office of Nuclear Reactor Regulation (NRR).

Are insignificant events really insignificant?

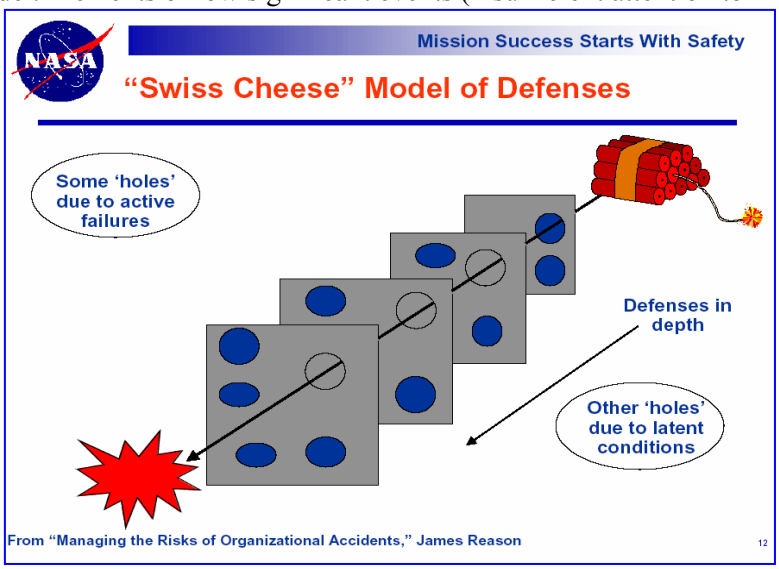
It is in this vein that the SCSS LER data may be used to systematically assess over 20 years' worth of commercial nuclear power plant operating experience data to identify adverse trends and recurring problems at domestic nuclear plants. The present ROP is structured to trend potential problem areas as negotiated with industry and with public input and feedback. Most operating experience documented in LERs is not individually significant enough to merit a ROP-initiated review. However, the data taken collectively may not only be informative but also important from a safety perspective. The collective data include human errors, management oversight issues, procedural problems, and equipment or system failures—the common building blocks of costly, avoidable, risk-significant events. It logically follows that if the sources of the minor problems are eliminated, the significant problems will also markedly decrease.

As illustrated by the Heinrich Pyramid, accidents arise from the same factors and conditions as less important incidents or even unreported occurrences. A focus only on important incidents or accidents is by definition a reactive rather than proactive approach—an after-the-fact assessment.



http://www.asy.faa.gov/gain/Presentation/sld007.htm

Visualized another way, accidents are almost always a breach of multiple defense-in-depth layers as illustrated in the "Swiss cheese" model. Elements of low significant events (insufficient attention to detail, time or economic pressures, uncorrected poor practices and habits, marginal maintenance or equipment care, etc.) are almost always present in significant events, whether these events take the form of reactor head degradation, precursors to nuclear plant core damage events, airline crashes, chemical plant explosions, or others involving complex systems and processes. Therefore, a reduction in the number of the "holes" in individual lines of defense (i.e., the cheese slices) gained through better knowledge or awareness of increasing trends in minor problems (each relatively insignificant by itself) and followed by appropriate action and attention may help prevent the "big" events.



Michael A. Greenfield, *The Inherent Values of Probabilistic Risk Assessment,* Second NASA PRA Workshop, June 19, 2001

SCSS quantifies recurring events

Both the Davis-Besse Lessons-Learned Report and the Kemeny Commission report noted that there were numerous precursors to the subject events. The SCSS can be used to examine LER operational data for adverse trends or recurring issues associated with the minor problems that have been shown to combine to cause major incidents or accidents if not identified and acted upon in a timely manner.

The SCSS is itself a product of the NRC actions taken in response to the TMI accident. NRC and ORNL staff designed SCSS to code, in a computer-searchable format, the sequence of occurrences described in LERs based on a thorough understanding of the event. SCSS specifically responds to a Kemeny Commission observation in the Overview section on Attitudes and Practices:

> *"A potentially insignificant incident grew into the TMI accident, with severe damage to the reactor. Since such combinations of minor equipment failures are likely to occur much more often than the huge accidents, they deserve extensive and thorough study."*

The SCSS coding captures in detail the combinations of human errors, component and piece-part failures, train- and system-level problems, their effects on unit operations (i.e, reactor trip, safety system actuation, etc.), and potential radioactive releases or personnel exposures. SCSS contains detailed data on over 49,000 operational events since 1980. It is available to NRC staff and contractors on a Web site (http://scss.ornl.gov) operated and maintained by ORNL for the NRC. The Web site reports approximately 2,500 hits per month in support of NRC staff research and evaluations.

The SCSS Web site permits NRC researchers to examine operational data for specific, individual problems (e.g., LERs reporting operator errors subsequent to a reactor trip from power operation or fouling of service water piping systems). These are typically retrospective or reactive types of searches.

Recognizing this perspective, **ORNL staff recently developed an objective and systematic methodology for examining the historical operational data proactively to identify potential problem areas before the occurrence of a significant event—an complementary and proactive approach to the present ROP data assessments. This approach "mines" the considerable operating experience data available in SCSS for issues potentially indicative of adverse trends and documents the comprehensive and systematic analyses of the data in a readily available Web format.** The approach will complement existing programs that examine individually significant events and address the issues and concerns raised by the authors of the Davis-Besse Lessons-Learned Report regarding inadequate or incomplete assessment and dissemination of reactor operating experience. An illustration of SCSS' use is provided below.

SCSS Recurring Event Application Overview

A two-step process is recommended for this review. Step 1 is to do structured queries of SCSS to quantify and review recurring operational events that frequently occur in combination to cause significant events. Step 2 is to document the review in a readily available and easy to use format. The query results and documentation could be shared with all NRC staff via the SCSS Web site, which is already well-known and in wide use across the agency. Step 1 is illustrated as follows.

Step 1 of the data analysis process is to use SCSS to identify recurring operational event sequences. A number of categories of recurring events were postulated, including:

- personnel activities and causes of those activities,
- component failures,
- train-level faults,
- system-level faults,
- ESF actuations,
- RPS actuations,
- LER reportability categories, and
- LER level causes.

The eight categories each have numerous subparts that combine to examine over 400 separate topics. For example, queries covered failures of components for over 30 different component classes, system level faults for over 150 systems, train level faults for over 150 systems, over 40 reportability reasons, and over 50 different personnel activity/personnel cause pairs.

Many of these categories of events represent the building blocks of significant events observed in LER data. Additional topic areas could be readily added (e.g., count system-component pairs to more specifically categorize component failures by system or to count systems affected by cascading event sequences). LERs often report (and SCSS captures) multiple, distinct problems and problems that affect other units at the site. In the analysis, each distinct occurrence of a problem was counted for each applicable unit. An examination of SCSS for individually significant events (e.g., reactor trips from power followed by failures of safety systems) was not included (although searchable in SCSS) because these events are examined or included in other NRC programs such as the Accident Sequence Precursor program or ROP indicators. However, any number of specific events or combinations could be readily added, if desired.

SCSS Recurring Event Example

To illustrate this process, the performance of the peer group of Babcock & Wilcox (B&W) plants was examined for LERs with event dates from January 1, 1999, through approximately August 31, 2002, for the eight search categories noted above. This analysis was performed for the peer group based upon the assumption that plants of a common design and vintage might be expected to experience and report similar types of problems, although any combination of plants could be used. Statistically significant differences in reporting may help focus limited staff review or analysis time to areas of meaningful differences. This approach identifies recurring issues that may be important collectively and also identifies where normal or expected reporting is not occurring. This could be due to exemplary performance or to different plant reporting thresholds or standards permitted by the NRC. The approach is to ensure that all operating experience is consistently reported and fully utilized, systematically and objectively.

The results of the queries are shown in Appendix A. Those data that were statistically above or below expected values for the peer group are highlighted. For the purpose of this example, a count was flagged when the probability of that count occurring randomly was significantly low or high compared to the peer group total count. Please note that the purpose of this analysis is to identify plants whose number of occurrences in a category was outside of expected patterns for the peer group, not to label a plant as "good" or "bad." Low counts do not always mean "good" and high counts do not automatically mean "bad."

In principle, each area of statistically high or low counts could be a focus of review activities. In practice, however, attention would likely be drawn to those areas that stand out more strongly. For example, Crystal River 3 consistently reported statistically significant low counts across all the categories and Oconee 1 and TMI 1 consistently reported statistically significant high counts. An illustration of the analysis process is shown in Appendix B.

Plant	Categories with High Counts	Categories with Low Counts
ANO 1	7	6
Crystal River 3	2	8
Davis-Besse	3	5
Oconee 1	12	1
Oconee 2	4	2
Oconee 3	3	3
TMI 1	22	2

Number of Category Outliers by Plant

Crystal River 3

Compared to the peer group, Crystal River 3 reported fewer problems with

- instrumentation and control (I&C) components,
- mechanical function components (typically, piece parts of pumps, valves, and other components),
- pipes and fittings,
- RHR trains,
- design errors, and
- reactor operations errors.

No failures were reported in many common problem areas. Good performance is noteworthy and should be shared so that others may learn and benefit. On the other hand, if the plant has different standards and practices for reporting than other similar plants, then guidance on reporting may be required to properly evaluate the plant's performance.

Oconee Units 1-3

The queries for Oconee 1 show 12 categories with high counts. A first level review shows a high number of reported problems with

- I&C components,
- pipes and fittings,
- control building HVAC trains,
- DC power trains,
- RHR trains and systems [Oconee Units 1-3 accounted for 7 of the 8 RHR problems in the peer group],
- RCS system problems,
- shutdowns or technical specification violations,
- automatic reactor trips [Oconee Units 1-3 accounted for 6 of the 7 automatic scrams in the peer group],
- design errors,
- task description errors (i.e., procedure-type problems), and
- reactor operations errors.

All of these potential problem areas may merit a follow-up review. However, a second-level review of only the task description problem area at Oconee was performed in order to illustrate the logical progression of the review process. Table 1 describes the observed problems based upon a brief LER review

The table shows that 13 LERs reported problems categorized as task description inadequacies, including problems with procedures, oral or written instructions, communications, or directions. Nine of the 13 LERs potentially affected all three units. One LER (266/99-007) reported multiple scenarios where guidance may not be adequate to effectively mitigate specific failures in the LPI and HPI systems, noting that "*there was no attempt within the Owners Group guidance or the [Oconee] site to address all potential failures within the EOP*" and that *"The issue of EOP adequacy is a recurring problem."* Six LERs involve the failure to thoroughly assess or evaluate the impacts of procedural changes.

Three Mile Island 1

TMI 1 reported a high number of problems relative to the peer group with

- electrical I&C function components (e.g., power supplies, circuit cards),
- filters,
- heat exchangers,
- I&C components,
- mechanical function components (typically, piece parts of pumps, valves, and other components),
- pipes and fittings,
- shock suppressors,
- structural function components,
- AFW system and trains,
- feedwater system and trains,
- reportability category concerning unanalyzed conditions,
- reportability category concerning events that could prevent fulfillment of a safety function,

- design errors, and
- task description errors (i.e., procedure-type problems).

TMI 1 reported a statistically high number of component failures in 10 of the 20 component classes. A follow-up review could be conducted on any of these areas as was done for Oconee.

Conclusion

Hundreds of NRC staff access the SCSS Web site each year for LER searches. In much the same way, the results of the operational experience queries and reviews could be made readily available to NRC staff. For example, each peer group could be evaluated on a periodic cycle (e.g., yearly, biannually, etc.). Results of the evaluation could be stored on the Web site for access and review. Both the query results tables with the statistical analyses and the details of specific data reviews could be made available. It is also possible for the user to request specific queries—groups of individual plants, time periods, etc.—that would be executed immediately.

This analysis process demonstrates the application of a systematic, objective, and comprehensive assessment of operating experience utilizing data recorded in SCSS focused on recurring problem areas. These are the building blocks of almost every significant event. This process provides a proactive perspective to operating experience assessment (i.e., identify and address potential problem areas before the occurrences of risk significant events) in contrast to the reactive nature of other significant event identification and analysis processes.

The process also provides a direct analysis tool to distill and more precisely locate identify LERs of interest as contrasted with manual reviews of hundreds or even thousands of LERs from less capable search engines. **It addresses valid concerns regarding problems with the assessment and dissemination of operating experience reviews raised over 20 years ago in the Kemeny Commission report—the reason for SCSS' creation—and echoed again in 2002 in the Lessons-Learned report on the Davis-Besse head corrosion problem.**

Table 1. Summary of Oconee Task Description Inadequacies.

LER	Errors/ LER	Activity	System Affected	Method of Detection	Comment	Units Affected
269/99-002	1	Admin	Low pressure injection	Review	Potential failure to establish LPI in 15min with certain single failures due to inadequate validation process	1,2,3
269/99-003	1	Admin	LTOP	Review	One train LTOP adminstrative controls deficient -- too short a time for operators to act under certain conditions due to misapplication of design input	1,2,3
269/99-005	1	Maintenance	Feedwater instrument lines	Operations abnormality	Misrouted feedwater instrumentation lines (for about 3 years) due to inadequate procedural guidance in maintaining adequate configuration control led to reactor trip	1
269/99-006	1	Test/calibration	Main steam pressure relief	Post trip review	New method for setting main steam relief valves introduced residual stresses causing valves to be out of tolerance low in actual use	1
269/99-007	4	Operations (4x)	LPI, HPI	Review	EOP and Abnormal Procedures may not be adequate to mitigate specific single failures in the LPI and HPI systems	1,2,3
269/99-008	1	Test/calibration	DC	Review	Maintenance supervisor did not notify operations as desired following discovery of a minor battery problem -- checkoff added to procedure	1
269/99-009	1	Admin	Steam generator	Review	Operation with unrepaired steam generator tube ends due to failure to recognize the need to propose a TS change request to exclude certain tube anomolies from SG tube repair criteria	1,2,3
269/00-002	1	Maintenance	Control room HVAC chilled water	Operations abnormality	All three units reduce power following loss of both trains of chilled water to the control room HVAC system due to insufficient procedural detail to tubing flare wear and subsequent leakage on both trains	1,2,3
269/00-004	1	Test/calibration	Emergency power	NRC notification	A more restrictive voltage and frequency limit imposed on the emergency power system was inadvertently included in the conversion to improved technical specifications	1,2,3
269/00-005	1	Test/calibration	Reactor protection	Testing	RPS main feedwater pump and main turbine trip setpoints slightly outside allowable values due to ineffective procedure review during the improved technical specification implementation process	1,2,3
269/00-007	1	Radiation protection	Radiation monitoring	Testing	Am-241 source lost due to not being secured when a test was interrupted	
269/02-001	1	Admin	Pressurizer	Testing	Pressurizer heaters powered from the standby shutdown facility inadequate to compensate for pressurizer heat transfer losses	1,2,3
269/02-004	1	Operations	LPI	Review	Inadvertant procedure change could cause overpressurization of alternate boron dilution flow path piping under certain conditions	1,2,3

Appendix A – Query Results and Statistical Analysis

Sequence Coding and Search System (SCSS) Searches for B&W Peer Group Outlier Performance

11-Nov-02

Outlier - High

Statistical Significance Level 0.1

Outlier - Low

No. of plants in peer group 7

Search for Component Failures Requiring Repair, Replacement, or Adjustment

Component Class	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
Batteries/Chargers	1				1				0.14
Blowers/Compressors	1							1	0.14
Circuit Breakers/Fuses	4	1				2		1	0.57
Electrical Conductors	4		1			3			0.57
Electrical/I&C Function Components	12	4			2	2		4	1.71
Equipment Interface Components	1							1	0.14
Filters Non-I&C	3							3	0.43
Heat Exchangers	14	1	1		4		1	7	2.00
Instrumentation & Control Components	62	2			15	16	13	16	8.86
Mechanical Function Components	50	14	3	6	1	6	3	17	7.14
Miscellaneous Components	10	1	2	1	1			5	1.43
Motors	3					1	1	1	0.43
Pipes/Fittings	80	10	1	3	18	4	24	20	11.43
Pumps	7	3	1					3	1.00
Shock Suppressors and Supports	9	2				1	2	4	1.29
Structural Function Components	10				2	2	2	4	1.43
Turbines	1	1							0.14
Valve Operators	1							1	0.14
Valves	40	1	4	17	7	4	2	5	5.71
Vessels	1			1					0.14

Actual or Potential Failures of Trains of Systems

Train of	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
Auxiliary Feedwater (PWR)	5			1				4	0.71
Chemical & Volume Control (PWR)	9				3	3	3		1.29
Chilled Water	2					2			0.29
Condensate and Feedwater	3		2		1				0.43
Control Building	2							2	0.29
Control Building HVAC	2				2				0.29
DC	3				3				0.43
Emergency Generator HVAC	1			1					0.14
Emergency Power Generation	2	1		1					0.29
Essential Raw Cooling/ Service Water	1				1				0.14
High Voltage AC (Greater Than 35kv)	1		1						0.14
Primary Coolant (PWR)	3	2				1			0.43
Reactor Building HVAC	4			2			2		0.57
Residual Heat Removal (PWR)	22			1	7	7	7		3.14
Steam Generator (PWR)	1	1							0.14

Actual or Potential System Level Failure

System	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
Auxiliary Feedwater (PWR)	3				1	1	1		0.43
Auxiliary Steam	1		1						0.14
Chemical & Volume Control (PWR)	4				1	2	1		0.57
Chilled Water	3				1	1	1		0.43
Circulating Water	1					1			0.14
Condensate and Feedwater	3		2		1				0.43
Control And Service Air	1			1					0.14
Control Building	3						1	2	0.43
Control Building HVAC	2				1			1	0.29
Emergency Generator Lube Oil	1	1							0.14
Essential Compressed Air	1			1					0.14
Fuel Building	2							2	0.29
Fuel Building HVAC	1	1							0.14
High Voltage AC (Greater Than 35kv)	1			1					0.14
Leak Monitoring	1			1					0.14
Nonnuclear Instrumentation	1			1					0.14
Pressurizer (PWR)	1			1					0.14
Primary Coolant (PWR)	8	1			3	2	1	1	1.14
Primary Reactor Containment (PWR)	4				1	1	1	1	0.57
Radiation Monitoring	1	1							0.14
Reactor Auxiliary Building	5				1	1	1	2	0.71
Reactor Building HVAC	1							1	0.14
Residual Heat Removal (PWR)	8	1			3	2	2		1.14
Secondary Containment HVAC-Standby Gas Treatment	1			1					0.14
Secondary Reactor Containment (PWR)	1			1					0.14
Spent Fuel Pool/ Refueling Pool Cooling and Cleanup	1			1					0.14
Turbogenerator	1						1		0.14
Turbogenerator Instrumentation and Control	1					1			0.14

Search of LERs by Reportability Code

Report Reason	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
--Special report, Part 21 report, etc.	1		1						0.14
10 CFR 20.2201(b)--Theft or loss of licensed material (original report).	1				1				0.14
10 CFR 20.2203(a)(3)(ii)--Excess radiation in an unrestricted area	1		1						0.14
10 CFR 50.46(a)(3)(ii)--Change or error in ECCS evaluation model.	3				1	1	1		0.43
10 CFR 50.73(a)(2)(i)(B)--Operation or condition prohibited by technical specifications.	25	1	4	5	3	4	5	3	3.57
10 CFR 50.73(a)(2)(i)--Shutdowns or technical specification violations.	44	5	4	7	12	6	8	2	6.29
10 CFR 50.73(a)(2)(ii)--Unanalyzed conditions.	37	2	3	1	6	6	7	12	5.29
10 CFR 50.73(a)(2)(iv)--ESF actuations.	18	5	3		3	4	1	2	2.57
10 CFR 50.73(a)(2)(v)(A)--Potential failure to shut down reactor and maintain safe condition.	3				1	1	1		0.43
10 CFR 50.73(a)(2)(v)(D)--Potential failure to mitigate accident consequences.	2	1			1				0.29
10 CFR 50.73(a)(2)(v)--Event that could have prevented fulfillment of a safety function.	5			2				3	0.71
10 CFR 50.73(a)(2)(vii)--Single failure criteria.	2			1				1	0.29
--Voluntary Submission	3	1						2	0.43

Search of LERs by LER-Level Cause Code

LER Level Cause	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
Communication Problem	3			2	1				0.43
Design Error or Inadequacy	41	4	1	2	8	8	8	10	5.86
Environmental Conditions	1		1						0.14
Equipment Failure	38	6	4	3	8	7	3	7	5.43
Human Error	35	4	2	7	5	5	5	7	5.00
Inadequate Training	1			1					0.14
Manufacturing Error or Inadequacy	4		1			1		2	0.57
Other Cause	1		1						0.14
Procedural Deficiency	34	5	5	8	5	4	4	3	4.86
Temperature	1				1				0.14
Unexpected Component Action or Response	1				1				0.14

Search of LERs by ESF Actuation Type

ESF Actuation	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
Automatic ESF Actuation--Electric Power	3			1				2	0.43
Automatic ESF Actuation--Flow	3	1	1		1				0.43
Automatic ESF Actuation--Level	2		1		1				0.29
Automatic ESF Actuation--Radiation	1	1							0.14
Total Automatic ESF	9	2	2	1	2			2	1.29

Search of LERs by Scram Type

Shutdown Type	Total	Arkansas Nuclear Unit #1	Crystal River Unit #3	Davis-Besse Unit #1	Oconee Unit #1	Oconee Unit #2	Oconee Unit #3	Three Mile Island Unit #1	Average
Automatic Scram	7	1			3	2	1		1.00
Manual Scram	2	2							0.29
Manual Shutdown	3		1		1	1			0.43
Unintentional Automatic Scram	1					1			0.14
Total Scrams/SDs	13	3	1		4	4	1		1.86

Search of LERs for Human Activity Errors									
Activity & Cause	**Total**	**Arkansas Nuclear Unit #1**	**Crystal River Unit #3**	**Davis-Besse Unit #1**	**Oconee Unit #1**	**Oconee Unit #2**	**Oconee Unit #3**	**Three Mile Island Unit #1**	**Average**
Administrative Activity--Intrinsic Human Error	14	1	3	3	4			3	2.00
Administrative Activity--Task Description Inadequacy	3			1				2	0.43
Design Activity--Intrinsic Human Error	26	4	1	2	7	1	1	10	3.71
Fabrication Activity--Intrinsic Human Error	5	2	1					2	0.71
Installation Activity--Intrinsic Human Error	2						1	1	0.29
Installation Activity--Task Description Inadequacy	1			1					0.14
Maintenance/ Repair Activity--Intrinsic Human Error	8	2	2	1			1	2	1.14
Maintenance/ Repair Activity--Task Description Inadequacy	14	5	1		2	1	1	4	2.00
Maintenance/ Repair Activity--Unknown Human Factor Cause	1							1	0.14
Operation Activity--Inadequate Man-machine Interface	3				1	1	1		0.43
Operation Activity--Intrinsic Human Error	6			4	1			1	0.86
Operation Activity--Proper Human Action or Response	4	2	1				1		0.57
Operation Activity--Task Description Inadequacy	18			4	5	4	4	1	2.57
Radiation Protection Activity--Intrinsic Human Error	2			1	1				0.29
Radiation Protection Activity--Proper Human Action or Response	1		1						0.14
Radiation Protection Activity--Task Description Inadequacy	2			1	1				0.29
Test/ Calibration Activity--Intrinsic Human Error	7	1		1	1	2	1	1	1.00
Test/ Calibration Activity--Proper Human Action or Response	1				1				0.14
Test/ Calibration Activity--Task Description Inadequacy	19	2	4	3	4	2	1	3	2.71
Unknown Activity--Task Description Inadequacy	1							1	0.14
Unknown Activity--Unknown Human Factor Cause	1							1	0.14
Sum Intrinsic Error	30	5	3	7	3	2	3	7	4.29
Sum Task Desc. Inad. plus Administration	71	8	8	13	16	7	6	13	10.14
Sum Maintenance	23	7	3	1	2	1	2	7	3.29
Sum Testing	29	5	4	4	6	4	2	4	4.14
Sum Rad. Protection	4	1		2	1				0.57
Sum Operations	25			3	7	5	5	5	3.57

Appendix B – Illustration of Analysis Process

Illustration of SCSS Recurring Event Analysis Process

Overview

The queries used in this analysis were designed to cover a broad range of operational event issues and potential problem areas not systematically covered, in many instances, in other review programs. They illustrate SCSS' flexibility and power. Search topics may be readily added or removed.

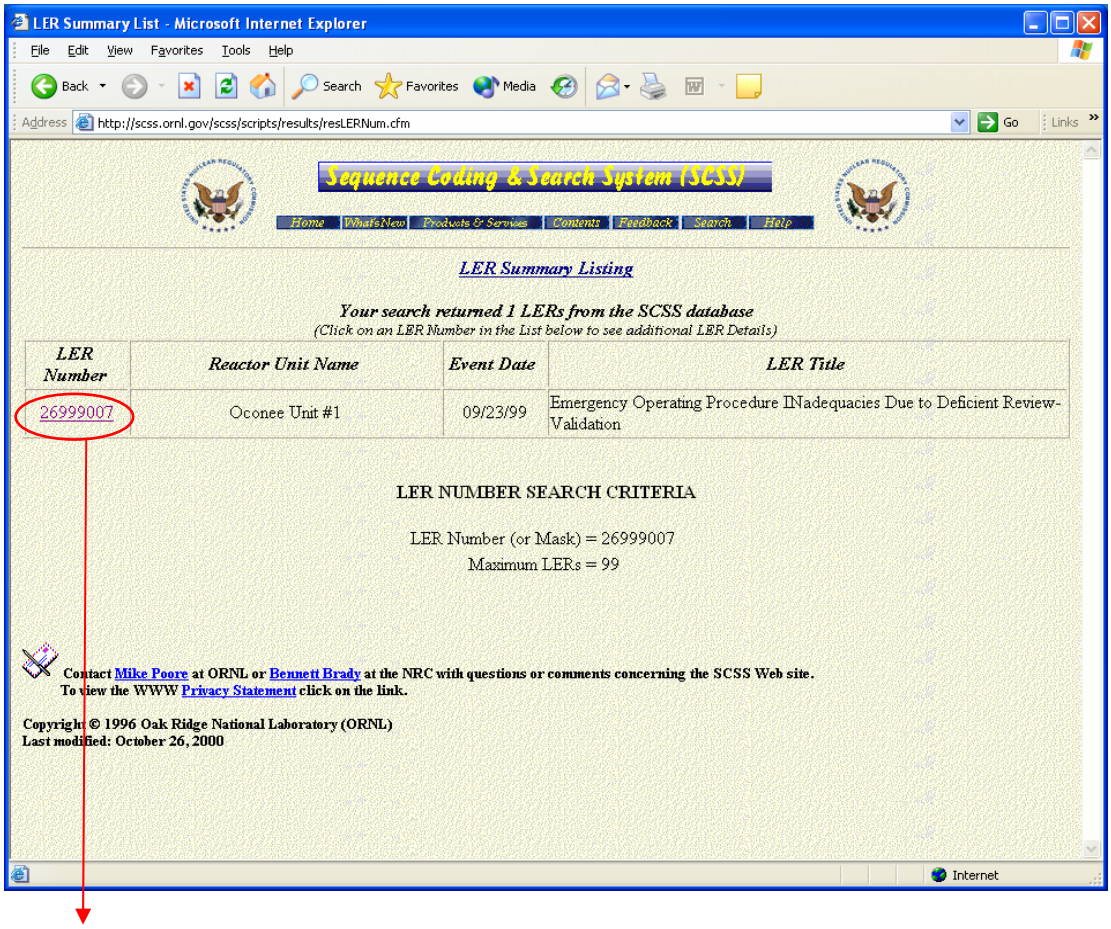
1. Determine the plants and time period of interest and execute the stored queries. A typical query result is shown below. Review the query results and determine those areas that are of interest or potentially merit further study.

Search of LERs for Human Activity Errors									
Activity & Cause	**Total**	**Arkansas Nuclear Unit #1**	**Crystal River Unit #3**	**Davis-Besse Unit #1**	**Oconee Unit #1**	**Oconee Unit #2**	**Oconee Unit #3**	**Three Mile Island Unit #1**	**Average**
Administrative Activity--Intrinsic Human Error	14	1	3	3	4			3	2.00
Administrative Activity--Task Description Inadequacy	3			1				2	0.43
Design Activity--Intrinsic Human Error	26	4	1	2	7	1	1	10	3.71
Fabrication Activity--Intrinsic Human Error	5	2	1					2	0.71
Installation Activity--Intrinsic Human Error	2						1	1	0.29
Installation Activity--Task Description Inadequacy	1			1					0.14
Maintenance/ Repair Activity--Intrinsic Human Error	8	2	2	1			1	2	1.14
Maintenance/ Repair Activity--Task Description Inadequacy	14	5	1		2	1	1	4	2.00
Maintenance/ Repair Activity--Unknown Human Factor Cause	1							1	0.14
Operation Activity--Inadequate Man-machine Interface	3				1	1	1		0.43
Operation Activity--Intrinsic Human Error	6			4	1			1	0.86
Operation Activity--Proper Human Action or Response	4	2	1				1		0.57
Operation Activity--Task Description Inadequacy	18			4	5	4	4	1	2.57
Radiation Protection Activity--Intrinsic Human Error	2			1	1				0.29
Radiation Protection Activity--Proper Human Action or Response	1		1						0.14
Radiation Protection Activity--Task Description Inadequacy	2			1	1				0.29
Test/ Calibration Activity--Intrinsic Human Error	7	1		1	1	2	1	1	1.00
Test/ Calibration Activity--Proper Human Action or Response	1				1				0.14
Test/ Calibration Activity--Task Description Inadequacy	19	2	4	3	4	2	1	3	2.71
Unknown Activity--Task Description Inadequacy	1							1	0.14
Unknown Activity--Unknown Human Factor Cause	1							1	0.14
Sum Intrinsic Error	30	5	3	7	3	2	3	7	4.29
Sum Task Desc. Inad. plus Administration	71	8	8	13	16	7	6	13	10.14
Sum Maintenance	23	7	3	1	2	1	2	7	3.29
Sum Testing	29	5	4	4	6	4	2	4	4.14
Sum Rad. Protection	4	1		2	1				0.57
Sum Operations	25			3	7	5	5	5	3.57

2. Click on the cell of interest to see the list of LERs and the coded events in those LERs that contribute to the cell count, in this case, task descriptions inadequacies or administrative errors.

PrGrpId	DOCKET	PlantName	Activity & Cause	Dscrmntr	LERNmbr	RprtblEvntDt
01	269	Oconee	Administrative Activity--Intrinsic Human Error	1	26999002	3/17/1999
01	269	Oconee	Administrative Activity--Intrinsic Human Error	1	26999003	5/12/1999
01	269	Oconee	Maintenance/ Repair Activity--Task Description Inadequacy	1	26999005	7/7/1999
01	269	Oconee	Test/ Calibration Activity--Task Description Inadequacy	1	26999006	8/18/1999
01	269	Oconee	Operation Activity--Task Description Inadequacy	1	26999007	9/23/1999
01	269	Oconee	Operation Activity--Task Description Inadequacy	4	26999007	9/23/1999
01	269	Oconee	Operation Activity--Task Description Inadequacy	2	26999007	9/23/1999
01	269	Oconee	Operation Activity--Task Description Inadequacy	3	26999007	9/23/1999
01	269	Oconee	Test/ Calibration Activity--Task Description Inadequacy	1	26999008	11/10/1999
01	269	Oconee	Administrative Activity--Intrinsic Human Error	1	26999009	11/15/1999
01	269	Oconee	Maintenance/ Repair Activity--Task Description Inadequacy	1	26900002	3/9/2000
01	269	Oconee	Test/ Calibration Activity--Task Description Inadequacy	1	26900004	9/6/2000
01	269	Oconee	Test/ Calibration Activity--Task Description Inadequacy	1	26900005	10/11/2000
01	269	Oconee	Radiation Protection Activity--Task Description Inadequacy	1	26900007	11/9/2000
01	269	Oconee	Operation Activity--Task Description Inadequacy	1	26902004	12/20/2001
01	269	Oconee	Administrative Activity--Intrinsic Human Error	1	26902001	3/7/2002

3. Click on the LER numbers to examine the individual LERs. In this example, LER 269/99-007 reported four separate task description inadequacies.



4. Click on the LER number to see the reference copy of the LER.

Duke Power.
A Duke Energy Company

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March 30, 2000

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Licensee Event Report 50/269/1999-07, Revision 1
Problem Investigation Process No.: O99-3123
O99-3702
O99-3703
O99-3863
O99-4113

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 269/1999-07, concerning guidance in the Emergency Operations Procedure (EOP) and associated Abnormal Procedures (APs) which may not be adequate to effectively mitigate specific single failures in the LPI and HPI systems. The inadequacies involve the expected time of failure discovery and the available time to take mitigating actions directed by the EOP. This supplement includes an additional example.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(ii). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

W. R. McCollum, Jr.

Attachment

IE22

[All pages of the LER are available: only the first page is shown here.]

Thus, out of the 49,000+ LERs currently on SCSS, we have gone with a few mouse clicks on an accessible Web site to a smaller set of events reporting human activity errors at B&W plants for a recent time period, to those with human errors associated with task description inadequacies and administrative activities for the peer group, to those for Oconee 1 as illustrated below:

49,000+ LERs on SCSS
→Approximately 160 human activity errors for the B&W peer group for the recent time period
→Approximately 70 task description or administrative problems for the peer group
→16 of these events at Oconee 1
→Text of the pertinent LERs displayed for review.