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Case Study: A Comparison of Generation Risk Assessment and Failure Modes and Effects Analysis Methodologies at TEPCO's Nuclear Power Plants

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Technical Update, June 2008

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CITATIONS

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This report describes research sponsored by the Electric Power Research Institute (EPRI) and Tokyo Electric Power Company (TEPCO).

This report is a corporate document that should be cited in the literature in the following manner:

Case Study: A Comparison of Generation Risk Assessment and Failure Modes and Effects Analysis Methodologies at TEPCO's Nuclear Power Plants. EPRI, Palo Alto, CA, and TEPCO, Tokyo, Japan: 2008. 1016461.

PRODUCT DESCRIPTION

This report describes research that compared the results of Generation Risk Assessment (GRA) and Strategic Reliability Centered Maintenance (S-RCM) methodologies. The report identifies similarities and differences obtained by the two methodologies and presents quantitative data to validate the qualitative results.

Results and Findings

Equipment importance classifications were similar using both GRA and S-RCM methodologies. The GRA methods ranked 48 components of a population of 492 equipment locations (approximately 10%) higher in importance than did the S-RCM techniques. Single-failure mechanisms were detectable earlier using GRA methods. Evidence of this effect was more dominant in the condensate system model; however, similar tendencies were seen in the turbine cooling water and seawater systems. Using the results of GRA from this study would lead to unnecessary maintenance on non-critical systems, structures, and components.

Challenges and Objectives

In Japan, the goal of developing a rational and safe maintenance strategy is being seriously discussed within the Japanese nuclear regulatory body. The application of risk information is being considered as one solution to realize this important goal. Therefore, it has been deemed necessary to reinforce the existing risk management skills that Tokyo Electric Power Company (TEPCO) personnel presently possess and apply them in the area of Nuclear Asset Management.

This technical update is the outcome of an Electric Power Research Institute (EPRI) supplemental offering for which TEPCO enlisted to model a pilot system using the GRA quantitative approach and contrast its results with those achieved using the S-RCM methodology on the same system.

Objectives of the study included identifying similarities and differences obtained by the two methodologies, validating the qualitative results with quantitative data, validating potential gaps between statistical risk models and failure modes and effects analysis (FMEA) qualitative experience models, and providing cost-benefit return on investment insights. In addition, validating the preventive maintenance program work accomplished thus far by using an alternative methodology would provide another level of certainty and credibility to the Japanese regulator when evaluating TEPCO maintenance strategies. This approach could prove beneficial to TEPCO and the broader Nuclear Asset Management community.

Application, Value, and Use

Many TEPCO plants have completed S-RCM analyses using FMEA techniques for balance-of-plant components; however, most have not performed risk based analyses for the same equipment. It remains to be seen whether TEPCO will consider this approach for their remaining plant systems after a credible cost-benefit analysis has been completed based on this effort at Fukushima Daiichi Unit 3 (1F3).

EPRI Perspective

GRA has been offered as a supplemental program as part of the overall Nuclear Asset Management program. Because most of the current GRA efforts in the United States were nearing completion, TEPCO was the sole subscriber to this supplemental program for 2007. TEPCO Systems Corporation (TEPSYS) was chosen to perform the GRA study.

TEPCO has been working with EPRI for the past three years to develop preventive maintenance programs at their three nuclear locations. The S-RCM approach provided a qualitative FMEA method to develop preventive maintenance tasks.

Approach

A project meeting was held, with TEPSYS in attendance. Initial project deliverables included several EPRI technical reports that were provided to TEPSYS before the meeting. These reports described EPRI-sponsored GRA research that had been completed to date. One of them, the EPRI report *Generation Risk Assessment (GRA) at Cooper Nuclear Station* (1011924), addressed the research that was conducted at Cooper Nuclear Station.

After reviewing these reports, TEPSYS expressed a desire to use the supplemental funding available to develop a GRA model for the same systems modeled at Cooper Nuclear Power Station, and to acquire the skills required to use their model to identify priorities and risks involved in performing maintenance at TEPCO's Fukushima Daiichi Nuclear Power Station. Under the proposed work scope, EPRI assisted TEPSYS in developing their GRA model and in acquiring the skills required to use the model.

Keywords

Generation Risk Assessment (GRA)
Failure Modes and Effects Analysis (FMEA)
Equipment Reliability
Strategic Reliability Centered Maintenance (S-RCM)

ABSTRACT

Tokyo Electric Power Company (TEPCO) has been working with the Electric Power Research Institute (EPRI) for the past three years to develop preventive maintenance programs at their three nuclear power generating facilities. The Strategic Reliability Centered Maintenance (S-RCM) approach has been used to provide a qualitative Failure Modes and Effects Analysis (FMEA) methodology to specify equipment functional importance (criticality) from which preventive maintenance tasks are developed.

This technical update is the outcome of an EPRI supplemental offering for which TEPCO enlisted to model a pilot system using the Generation Risk Assessment (GRA) quantitative approach and contrast its results with those achieved using the S-RCM methodology on the same system. Objectives of the study included identifying similarities and differences obtained by the two methodologies, validating the qualitative results with quantitative data, validating potential gaps between statistical risk models and FMEA qualitative experience models, and providing cost-benefit return on investment insights. In addition, validating the preventive maintenance program work accomplished thus far by using an alternative methodology would provide another level of certainty and credibility to the Japanese regulator when evaluating TEPCO maintenance strategies. This approach could prove mutually beneficial to TEPCO and the broader Nuclear Asset Management community.

Many TEPCO plants have completed S-RCM analyses using FMEA techniques for balance-of-plant components; however, most have not performed risk based analyses for the same equipment. It remains to be seen whether TEPCO will consider this approach for their remaining plant systems after a credible cost-benefit analysis has been completed based on this effort at Fukushima Daiichi Unit 3 (1F3).

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1
INTRODUCTION

1.1 Background

In Japan, the goal of developing a rational and safe maintenance strategy is being seriously discussed within the Japanese nuclear regulatory body. The application of risk information is being considered as one solution to realize this important goal. Therefore, it has been deemed desirable to reinforce the existing risk management skills that Tokyo Electric Power Company (TEPCO) personnel presently possess and apply them in the area of Nuclear Asset Management.

Generation Risk Assessment (GRA) has been offered as a supplemental program as part of the overall EPRI Nuclear Asset Management program. Because most of the current GRA efforts in the United States were nearing completion, TEPCO was the sole subscriber to this supplemental program for 2007.

A project meeting was held, with TEPSYS in attendance. Initial project deliverables included several Electric Power Research Institute (EPRI) technical reports that were provided to TEPSYS before the meeting. These reports described EPRI-sponsored GRA research that had been completed to date. One of them, the EPRI report *Generation Risk Assessment (GRA) at Cooper Nuclear Station* (1011924), addressed the research that was conducted at Cooper Nuclear Station. After reviewing these reports, TEPSYS expressed a desire to use the supplemental funding available to develop a GRA model for the same systems modeled at Cooper Nuclear Power Station, to develop their own model based on this model, and to acquire the skills required to use their model to identify priorities and risks involved in performing maintenance at TEPCO's Fukushima Daiichi Nuclear Power Station. Under the proposed work scope, EPRI assisted TEPSYS in developing their GRA model and in acquiring the skills required to use the model.

TEPCO has been working with EPRI for the past three years to develop preventive maintenance programs at their three nuclear locations. The Strategic Reliability Centered Maintenance (S-RCM) approach provided a qualitative failure modes and effects analysis (FMEA) method to develop preventive maintenance tasks.

EPRI suggested that an alternative GRA project approach, if used by TEPCO, would model a pilot system using GRA methodology and leveraging the S-RCM work completed to date to validate the qualitative results with quantitative data. Additional potential outcomes could include validating potential gaps between statistical risk models and FMEA qualitative

experience models, and providing cost-benefit return on investment insights. In addition, validating the preventive maintenance program work accomplished thus far by using an alternative methodology would provide another level of certainty and credibility to the Japanese regulator when evaluating TEPCO maintenance strategies. This approach could prove mutually beneficial to TEPCO and the broader Nuclear Asset Management program community.

1.2 Objectives

Objectives of the study included identifying similarities and differences obtained by the two methodologies, validating the qualitative results with quantitative data, validating potential gaps between statistical risk models and FMEA qualitative experience models, and providing cost-benefit return on investment insights. In addition, validating preventive maintenance program work accomplished thus far by using an alternative methodology would provide another level of certainty and credibility to the Japanese regulator when evaluating TEPCO maintenance strategies.

2
DISCUSSION OF GENERATION RISK ASSESSMENT MODEL

According to the EPRI report *Generation Risk Assessment (GRA) Plant Implementation Guide* (1008121), the first step in the GRA process is to select the systems for modeling. The selected systems should have, or be perceived to have, an important impact on the generation capability of the plant while also being candidates for plant operating or maintenance changes that can be assessed using the GRA results as input.

Table 2-1 compares the systems selected for modeling at the Cooper Nuclear Power Station to those selected for GRA modeling by TEPCO at their Fukushima Daiichi Unit 3 (1F3) as well as the systems that had been previously analyzed at 1F3 using the qualitative S-RCM methodology.

For the 1F3 GRA, it was decided to select systems of high importance that had previously been analyzed by TEPCO and EPRI using the qualitative S-RCM methodology and had also been modeled at Cooper Nuclear Power Station. This allowed the comparison of 1F3 modeling results directly to the results of those same systems previously analyzed at Cooper Nuclear Power Station as well as a validation of the previously obtained qualitative S-RCM results for the same systems at 1F3.

The following systems were chosen for GRA modeling at 1F3:

- Feedwater
- Condensate water
- Circulating water
- Switchyard
- Instrument Air

Table 2-2 illustrates the process used to develop the fault trees for the Instrument Air System. This table identifies functional failures in various operating conditions that result in the generation effects being modeled. The results of Table 2-2 reflect an independent failure analysis similar to the FMEA performed in the S-RCM database. Thus, the fault trees generated from this analysis provide an independent comparison with the S-RCM results originally developed at 1F3. The following subsections summarize the modeling results of these systems.

Table 2-1
Results of selection of modeling systems

Systems That Can Cause Megawatt-Hour Loss	Strategic Reliability Centered Maintenance (high importance)	Cooper Nuclear Power Station Model	TEPCO Generation Risk Assessment Model (1F3)
Reactor recirculation system			
Nuclear boiler system			
Control rod drive system			
Neutron system			
Clean-up system			
Atmospheric control system			
Fuel pool cooling system			
High-pressure core injection system			
Valve grand leak treatment system			
Turbine lube-oil system	X		
Steam turbine	X		
Condensate demineralizer system			
Feedwater system	X	X	X
Condensate water system	X	X	X
Heater drain system	X		
Reactor cooling water system			
Turbine cooling water system			
Auxiliary seawater system		X	
Circulating water system	X	X	X
Power system	X		
Switchyard	X	X	X
Transformer	X		
Generator		X	
Instrument air system	X	X	X
Service air system			
House boiler			
Fire protection system			
Makeup water system	X		
H_2/O_2 supply system			

Table 2-2
Root cause analysis (Instrument Air System)

Equipment ID	Top event	1st cause	2nd cause	3rd cause	4th cause	5th cause	6th cause	7th cause	8th cause	9th cause	10th cause	11th cause
COMP-71-100AB	IA system failure (100% derate)	Front line failure	IA system failure AND SA system failure AND Adjacent power plant SA failure	IA system failure	IA compressor failure	Pump fail to run (CCF)						
COMP-71-100A						Independent failure	"A" system failure AND "B" system failure	"A "system failure	"A "system compressor failure	Pump fail to run		
RV-71-230A									Safety valve premature open			
V-71-252A									Manual valve plugging			
V-36-12-15									TCW line plugging in IA system A	Manual valve plugging		
V-36-12-17										Manual valve plugging		
V-36-61A										Manual valve plugging		
V-36-62A										Manual valve plugging		
V-36-65A										Manual valve plugging		
V-36-66A										Manual valve plugging		
SV-36-60A										Solenoid Valve close without command		

Table 2-2 (continued)
Root cause analysis (Instrument Air System)

Equipment ID	Top event	1st cause	2nd cause	3rd cause	4th cause	5th cause	6th cause	7th cause	8th cause	9th cause	10th cause	11th cause
TCV-36-64A										Air operated valve close without command		
STR3SB									IA-A system Power loss	MC3C loss	Transformer failure	Transformer/Current transformer loss of function
MC3SB1											Breaker failure	Breaker premature open
MC3SB8												Breaker premature open
MC3B3												Breaker premature open
MC3C1												Breaker premature open
MC3B											Bus loss of function	Bus loss of function
MC3SB												Bus loss of function
MC3C												Bus loss of function
PC3C4B										PC3C failure	Breaker failure	Breaker premature open
MC3C12												Breaker premature open
TRPC3C											Transformer/Current transformer loss of function	
PC3C											Bus loss of function	

Table 2-2 (continued)
Root cause analysis (Instrument Air System)

Equipment ID	Top event	1st cause	2nd cause	3rd cause	4th cause	5th cause	6th cause	7th cause	8th cause	9th cause	10th cause	11th cause
PC3C7B										Breaker premature open		
TBMCC3C2										Bus loss of function		
TBMCC3C2-2E										Breaker premature open		
COMP-71-100B								"B" system failure	"B "system compressor failure	Pump fail to run		
COMP-71-100B										Pump fail to start		
RV-71-230B									Safety valve premature open			
V-71-252B									Manual valve plugging			
V-71-251B									Solenoid Valve fail to open			
V-36-12-16									TCW line plugging in IA system B	Manual valves plugging		
V-36-12-18										Manual valves plugging		
V-36-61B										Manual valves plugging		
V-36-62B										Manual valves plugging		
V-36-65B										Manual valves plugging		
V-36-66B										Manual valves plugging		

Table 2-2 (continued)
Root cause analysis (Instrument Air System)

Equipment ID	Top event	1st cause	2nd cause	3rd cause	4th cause	5th cause	6th cause	7th cause	8th cause	9th cause	10th cause	11th cause
SV-36-60B										Solenoid Valve close without command		
TCV-36-64B										Air operated valve close without command		
STR3SB									IA-B system Power loss	MC3D loss	Transformer failure	Transformer/Current transformer loss of function
MC3SB1											Breaker failure	Breaker premature open
MC3SB8												Breaker premature open
MC3B3												Breaker premature open
MC3D1												Breaker premature open
MC3B											Bus loss of function	Bus loss of function
MC3SB												Bus loss of function
MC3D												Bus loss of function
PC3D4B										PC3D failure	Breaker failure	Breaker premature open
MC3D12												Breaker premature open
TRPC3D											Transformer/Current transformer loss of function	

Table 2-2 (continued)
Root cause analysis (Instrument Air System)

Equipment ID	Top event	1st cause	2nd cause	3rd cause	4th cause	5th cause	6th cause	7th cause	8th cause	9th cause	10th cause	11th cause
PC3D											Bus loss of function	
PC3D7B										Breaker premature open		
TBMCC3D2										Bus loss of function		
TBMCC3D2-3E										Breaker premature open		

2.1 Feedwater System

The results from modeling the Feedwater System are the following:

- 100% derate
  - Failure of Feedwater or Condensate System valves.
  - Failure of all (two of two) Turbine-Driven Reactor Feed Pumps.
  - Failure of one Turbine-Driven Reactor Feed Pump and all (two of two) Motor-Driven Reactor Feed Pumps.
  - Failure of any support system (such as Lube Oil, Service Water, Closed Cooling Water, Instrument Air, Electrical Power, or Control).
- 66% derate
  - Failure of one (one of two) Turbine-Driven Reactor Feed Pumps and failure of one (one of two) Motor-Driven Reactor Feed Pumps. In this scenario, Feedwater flow is reduced to 75%. When the Reactor water level decreases and reaches L-4, a primary loop recirculation pump run-back occurs and plant power is reduced to approximately 70%.

2.2 Condensate Water System

The Condensate System was modeled with the Feedwater System.

2.3 Circulating Water System

The results from modeling the Circulating Water System are the following:

- 100% derates
  - Failure of the Circulating Water System results in reduced power and eventual shutdown due to increased Turbine exhaust backpressure from loss of Condenser vacuum, decreased Condenser Hotwell level, and increased Turbine exhaust hood temperatures.
  - Failure of all (three of three) Circulating Water System trains.
  - Failure of one (one of three) Circulating Water Pump or Motor. In this scenario, the pump must be re-centered after repairs have been completed, which requires underwater access to the pump. This cannot be accomplished with the remaining two pumps in operation; therefore, the plant must be shut down.
- Derates less than 100%
  - Failure of two (two of three) trains (66% derate).
  - Failure of one (one of three) train (33% derate).
  - Failure of a Backwash Water valve on one Main Condenser Waterbox (10% derate).

2.4 Switchyard

The results from modeling the Switchyard System are the following:

- 100% derates. Failure of Switchyard components requires plant shutdown; therefore, these scenarios are modeled as 100% derates.
- Plant derates less than 100% are not considered in this assessment.

2.5 Instrument Air System

The results from modeling the Instrument Air System are the following:

- 100% derates. Main header pressure is maintained at approximately 0.7 MPa. Failures that result in a pressure drop in the main air header to less than 0.44 MPa result in a 100% derate condition. Consequently, failure of any component that results in the inability to provide compressed air at a rate that is adequate to make up for expected air header leakage results in a plant trip and a 100% derate condition.
- Plant derates less than 100% are not considered in this assessment.

3
COMPARISON OF STRATEGIC RELIABILITY CENTERED MAINTENANCE RESULTS WITH GENERATION RISK ASSESSMENT MODELING

3.1 Strategic Reliability Centered Maintenance Work Performed at Fukushima Daiichi

TEPCO has been using S-RCM analysis as a process to enhance nuclear safety and reliability of Systems, Structures, and Components. Through this qualitative process, they have developed procedures for planning maintenance objects, defining equipment importance, and defining maintenance type.

In particular, the decision process for determining equipment importance is based on the following criteria:

- Impact on plant when the equipment failed
- Failure probability for the equipment
- Detectability of the equipment failure

TEPCO has performed a qualitative evaluation of equipment importance based on these criteria. The equipment importance determination process is described in the following subsections.

3.1.1 Definition of Equipment Importance

Table 3-1 provides definitions of the four S-RCM categories of equipment importance: Critical 1, Critical 2, Non-Critical, and Run to Failure.

Table 3-1
Definition of equipment importance

Equipment Importance	Definition	Description
Critical 1	Most significant equipment	Much greater impact on plant. Objective of maximum preventive maintenance.
Critical 2	Significant equipment	Greater impact on plant. Objective of prioritized maintenance.
Non-Critical	Preventive maintenance equipment	Based on cost and impact on the plant, preventive maintenance, rather than corrective maintenance, is appropriate.
Run to Failure	Corrective maintenance equipment	Corrective maintenance.

3.1.2 Criteria for Determining Equipment Importance

The decision process for determining equipment importance is based on the following criteria:

- Impact on plant (IN) when the equipment failed; evaluate for safety, supply reliability, and cost
- Failure probability (FP) for the equipment; determine equipment failure rate for the past 10 years
- Detectability (DE) of the equipment failure; determine how readily equipment degradation can be detected before failure

These three criteria are used to determine the equipment importance category for each component. The highest level was categorized as 4, and the lowest level was categorized as 1.

3.1.3 Determination of Equipment Importance Number

The equipment importance number (EIN) is calculated as follows:

$$EIN = IN \times FP \times DE$$

The resulting product determines the equipment importance level, as follows:

- Critical 1: EIN >20.
- Critical 2: EIN >10 but <20 and IN >4. (At 1F3, the S-RCM analysis did not discriminate between Critical 1 and Critical 2 because maintenance tasks were essentially identical.)
- Non-Critical: EIN >10 but <20, and IN <4, and “yes” is the answer for at least one of the four questions*.
- Run to Failure: EIN <20, and IN <4, and “no” is the answer for all four questions*.

* Four questions:

1. Is there failure history for the equipment?
2. Is the equipment failure high-impact?
3. Is the task of preventive maintenance easy for the equipment?
4. Is the cost of preventive maintenance lower than the cost of corrective maintenance?

Figure 3-1 illustrates an example of an S-RCM evaluation result report for a Circulating Water Pump.

S-RCM Evaluation Result Report | Turbine G | RCM No.100 | Last update | 1/30/2006

1F3	RCM System	038 Circulating Water System	Type	Pump

Equipment No	38-3A_P
Equipment name	Circulating Water Pump A

Group Equipment

38-3B_P, 38-3C_P

Equipment Specification

Type	: Pump
Capacity	: 884m3/min
Total Head	: 10.5m
Body Material	: Stainless steel JIS SCS14
Impeller Material	: Stainless steel JIS SCS14
Shaft Material	: S35C+SUS32
RPM	: 720rpm

Regulation

N/A

Redundancy

N/A

Spare parts

N/A

Equipment Function

【Water Conveyance Function】
Function to supply seawater to condensers

○Interlock
This pump start up with cooling water normal capacity (70L/Min) and more than 20% valve opening of discharge valve. (The discharge valve automatically open still 20% with pump CS "ON".)

【Boundary Function】

Importance

1 【Impact on Plant:4 Failure Probability:4 Detectability:4】 (Importance: 1 or 2 → Critical, 3 or 4 → Non-Critical)

【CHS】(Critical : C, Frequency of Use : H, Environment : S)

Basis of Importance

【Water Conveyance Function】

○Loss of Function in Operation
【Impact on Plant】 Plant power down. Strain operators. (4 Point)
【Failure Probability】 Maintenance has been performed per several year because the pump is corroded by seawater and sand in sea. (3 Point)
【Detectability】 Monitoring by thermography, lubricating oil diagnostics and vibration diagnostics is difficult because equipments are submerged in intake channel. However,
leakage of the shaft seal part can be checked. (4 Point)

○Loss of function before start
【Impact on Plant】 The plant can't start up still completing repair.(4 Point)
【Failure Probability】 Maintenance has been performed per several year because the pump is corroded by seawater and sand in sea. (3 Point)
【Detectability】 Monitoring by thermography, lubricating oil diagnostics and vibration diagnostics is difficult because equipments are submerged in intake channel. However,
leakage of the shaft seal part can be checked. (4 Point)

【Loss of boundary function】
【Impact on Plant】 Gland leakage result in plant shutdown or power down. (4 Point)
【Failure Probability】 If maintenance has not been performed, leakage is actualized within a few year. (4 Point)
【Detectability】 Leakage of the shaft seal part can be checked. However, if an "Exposed thread" is lost, it is difficult to repair without plant power down. (Point 4)

The "Importance" is 【Critical】"1", the reasons are (1) no-mitigating methods and (2) impact on power.

Effect in Loss of Function

【Loss of Water Conveyance Function】
(Loss of the Function in Operation)
「Accidental Operational procedure (Event Base) Section 10, Turbine System Accident subsection 5, Circulating water pump 1 pump trip 」

Continuous operation of rated power become difficult by vacuum down, hotwell water level down and low pressure turbine exhaust room temperature raise due to loss of condensate function. (Plant power down.)

Operation of Operators
If turbine generator monitoring parameter shows significant fluctuation, immediately power down. Then operators monitor condenser vacuum and power down so that the pressure keep less than 100mmHgabs. And operators should monitor that solenoid valve open

The operator recognize that the circulating water system or intake system is abnormal, by warning signal of "CW pump failure" of "Intake structure failure".

(Loss of the function before start)
On line maintenance is impossible due to intake stream. Therefore, outage is extended.

【Loss of boundary function】
Small leakage is not important, because components are submerged in intake channel. However, large leakage affect the pump operation (because large leakage result in scattering of water in the pump room).

Q1, Is there failure histories? 「Y」	Q2, Are there strain operators? 「Y」	Q3, Is there simple task? 「N/A」	Q4, Is it low cost to continue preventive maintenance. 「Y」
For maintenance, corrosion with seawater and line plugging is found.	It is necessary that power down operation following "Accidental Operation Procedure".		Preventive maintenance is lower costs than the corrective maintenance costs.

Task

Actual Inspection	Actual Period	Incentive Inspection	Period	Remarks
Patrol by duty operator	Everyday	Patrol by duty operator		
		Vibration measurement	Per 3 Month	Measured only topmost BRG.
		SE Patrol	Per 6 Month	
Full Inspection	Per 3 Periodic Inspection	Full Inspection	Per 3 Periodic Inspection	Shaft seal part・Lubricating water line, C/R, Test run
Simplified Inspection	Every Periodic Inspection	Simplified Inspection	Every Periodic Inspection	Only test run

Validity of Maintenance

○The pump is maintenance during plant outage, because the pump can't be repaired by on-line and single failure of the pump is effected on power.
○It is reported a lot that failure caused by corrosion with seawater. It is necessary that inside of the p

Remarks

P&ID SH-87,88

Figure 3-1
Example of S-RCM evaluation report

3.2 Description of Four-Quadrant Plot

An equipment importance evaluation was performed for the TEPCO 100% derate GRA model. The risk importance index was based on the Fussell-Vesely (FV) and Risk Achievement Worth (RAW) risk parameters. Figures 3-2 and 3-3 illustrate FV and RAW plots for the top 100 equipment locations included in this model at 100% plant derate. Figure 3-4 represents the Four-Quadrant plot of the model.

Figure 3-2 demonstrates that the Circulating Water Pumps and Seal Drain Collector Pumps are the most significant from the perspective of this measure of risk. When the Circulating Water Pump has failed, it requires re-centering after repairs have been completed. On-line maintenance will not be attempted because underwater access to the pump inlet is required to re-center the pump; this requires that the plant be shut down for repairs to be completed. Failure of a Seal Drain Collector has no impact on plant. However, the pump cannot be properly isolated to make repairs on-line; therefore, a plant generation shutdown is required to repair it.

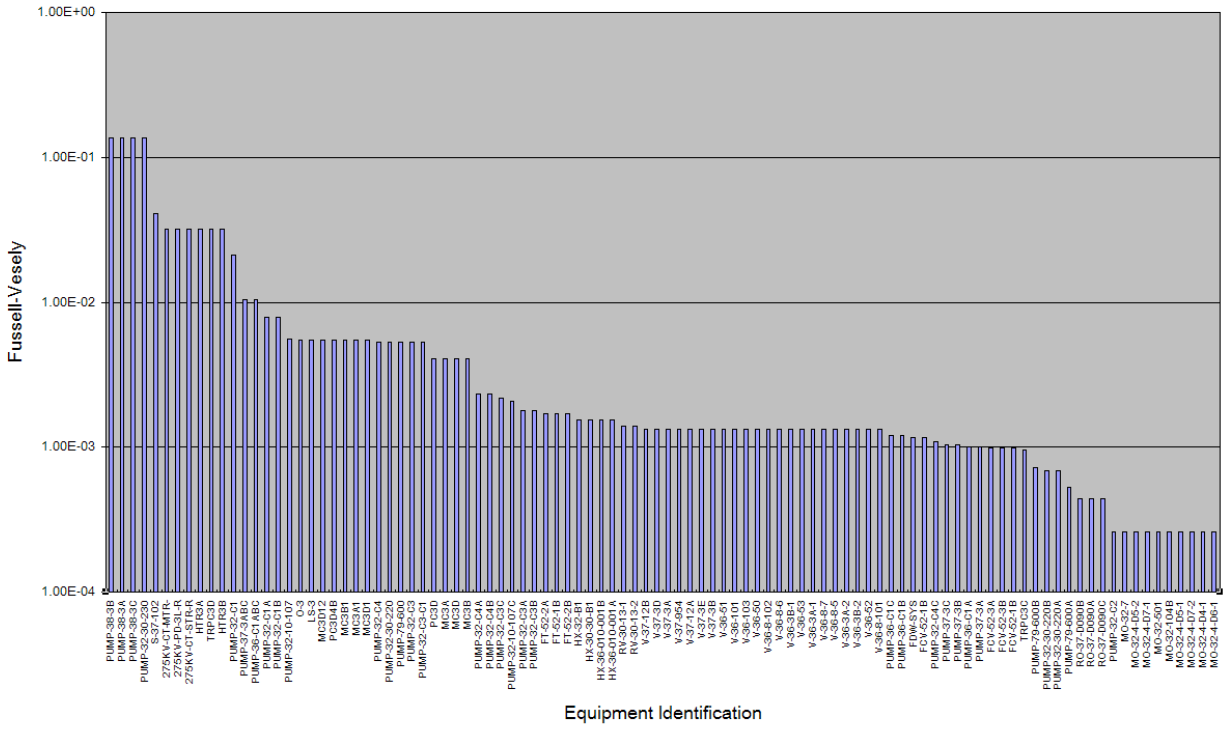

Figure 3-2
Fussell-Vesely at 100% derate (top 100 equipment locations)

Figure 3-3, on the other hand, shows that the RAW for the top 100 equipment locations have the same value. The results are similar to those in the EPRI report *Generation Risk Assessment (GRA) at Cooper Nuclear Station* (1011924), in which the RAW values were essentially the same for any given derate level.

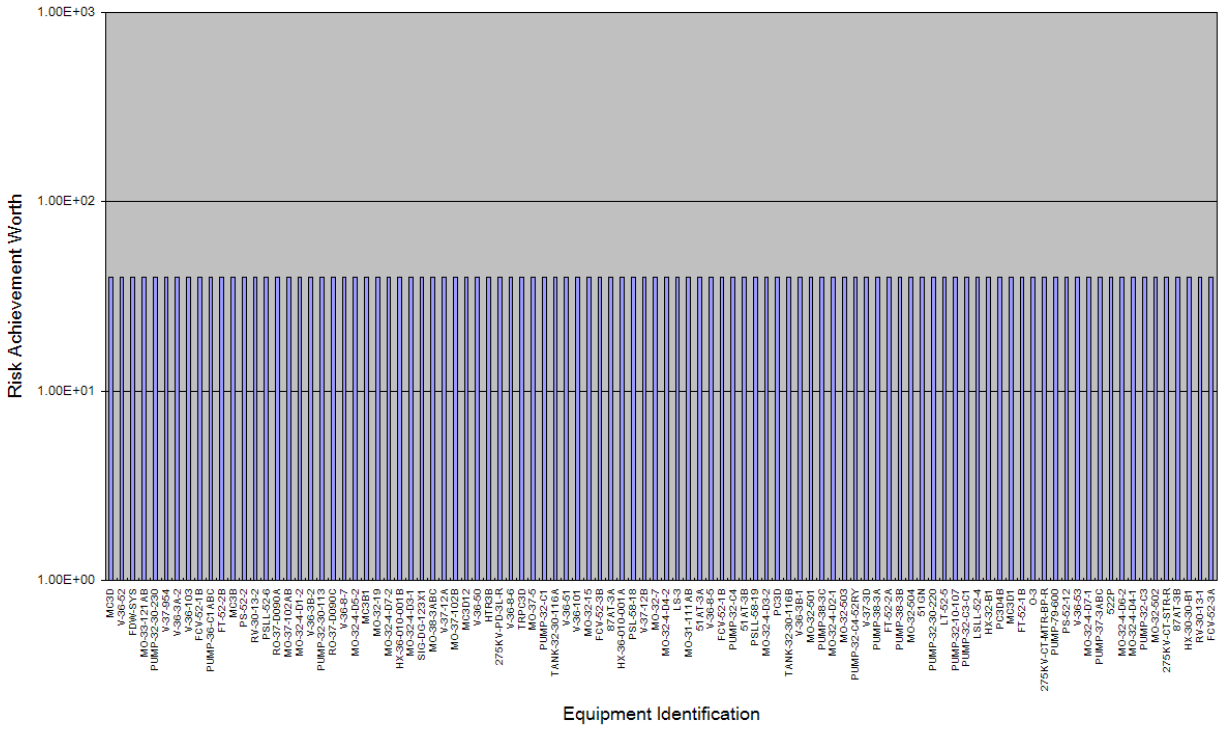

Figure 3-3
Risk Achievement Worth at 100% derate (top 100 equipment locations)

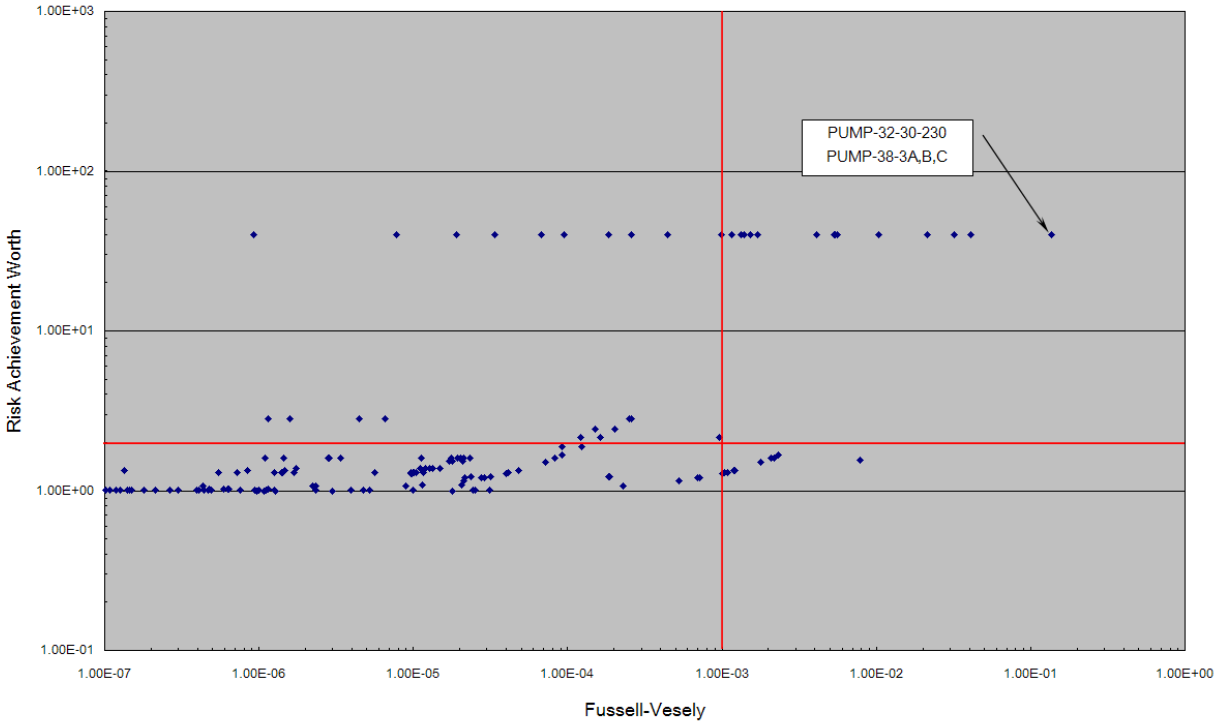

Figure 3-4
Fussell–Vesely versus Risk Achievement Worth at 100% derate

3.3 Evaluation of the Differences

3.3.1 Purpose

The purpose of this section is to demonstrate the validity of the qualitative equipment importance classification resulting from the TEPCO S-RCM by comparing it to the quantitative equipment importance classification determined as a result of GRA modeling.

3.3.2 Equipment Importance Classification by Generation Risk Assessment

The risk importance index was based on the FV and RAW risk parameters.

As with S-RCM, the importance classifications were categorized into four levels. The values defined were used in the TEPCO maintenance program as the threshold value. Figure 3-5 illustrates the following equipment importance classification arrived at by GRA modeling:

- Critical 1: FV ≥ 0.001 and RAW ≥ 2.0.
- Critical 2: FV ≥ 0.001 or RAW ≥ 2.0.
- Non-Critical: FV < 0.001 and RAW < 2.0.
- Run to Failure: Not modeled in GRA. (This philosophy reflects conservatism that might not be cost effective due to unnecessary maintenance on Non-Critical equipment.)

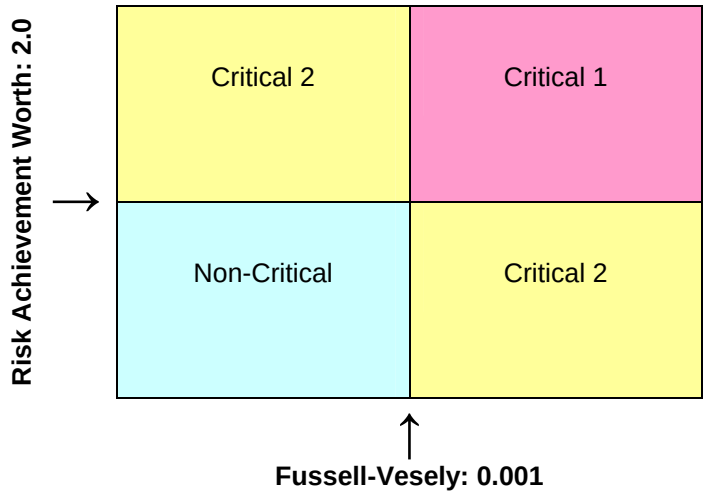

Figure 3-5
TEPCO Generation Risk Assessment equipment importance classification matrix

3.3.3 Evaluation of the Differences

Figure 3-6 illustrates a comparison of the equipment importance classifications from the S-RCM and GRA methodologies.

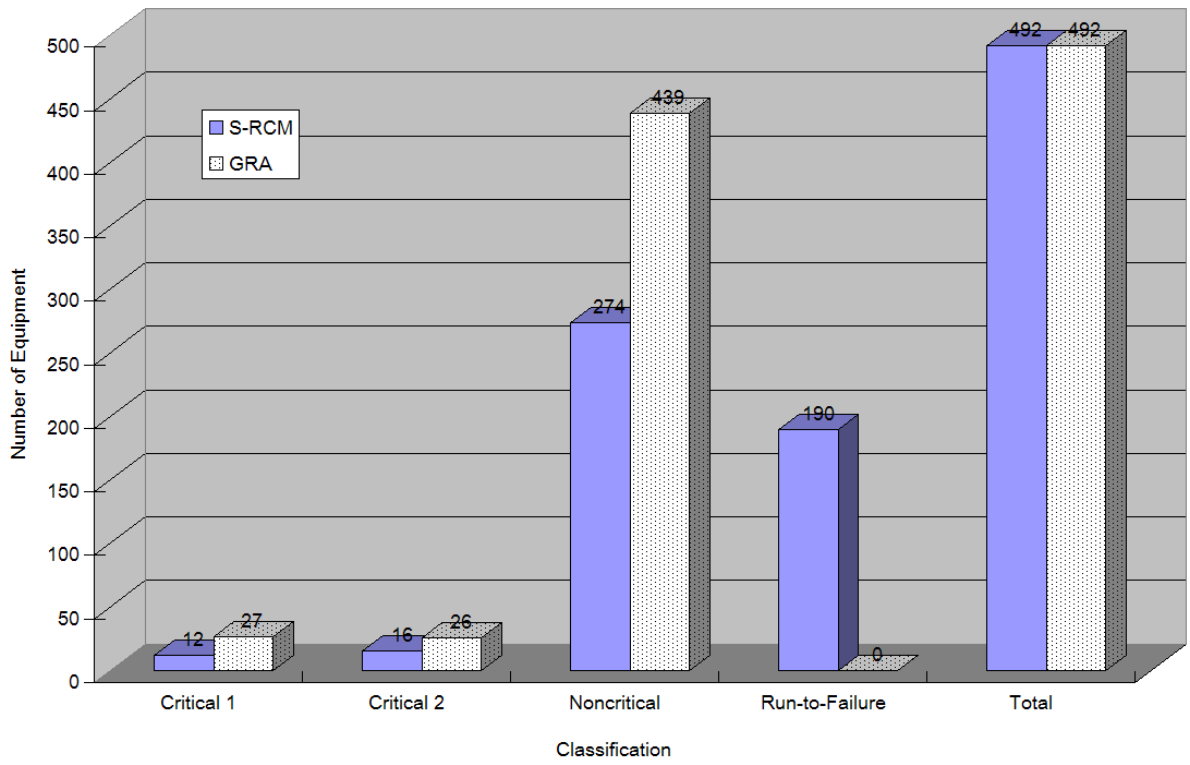

Figure 3-6
Distribution of equipment by importance classification using Strategic Reliability Centered Maintenance and Generation Risk Assessment mmethods

The S-RCM and GRA modeling importance evaluations were performed on 492 equipment locations. The S-RCM evaluation resulted in 28 being rated Critical 1 or Critical 2, whereas the GRA modeling resulted in 53 locations being rated Critical 1 or Critical 2.

Figure 3-7 illustrates the critical 1 and 2 equipment classifications by system. For Feedwater, S-RCM identified 17 equipment locations as Critical 1 or 2, whereas GRA identified only 12. From analysis of the specific equipment locations, it can be concluded that important equipment locations are differences in the identified with respect to their impact on plant startup and outage and detectability of equipment failure in S-RCM.

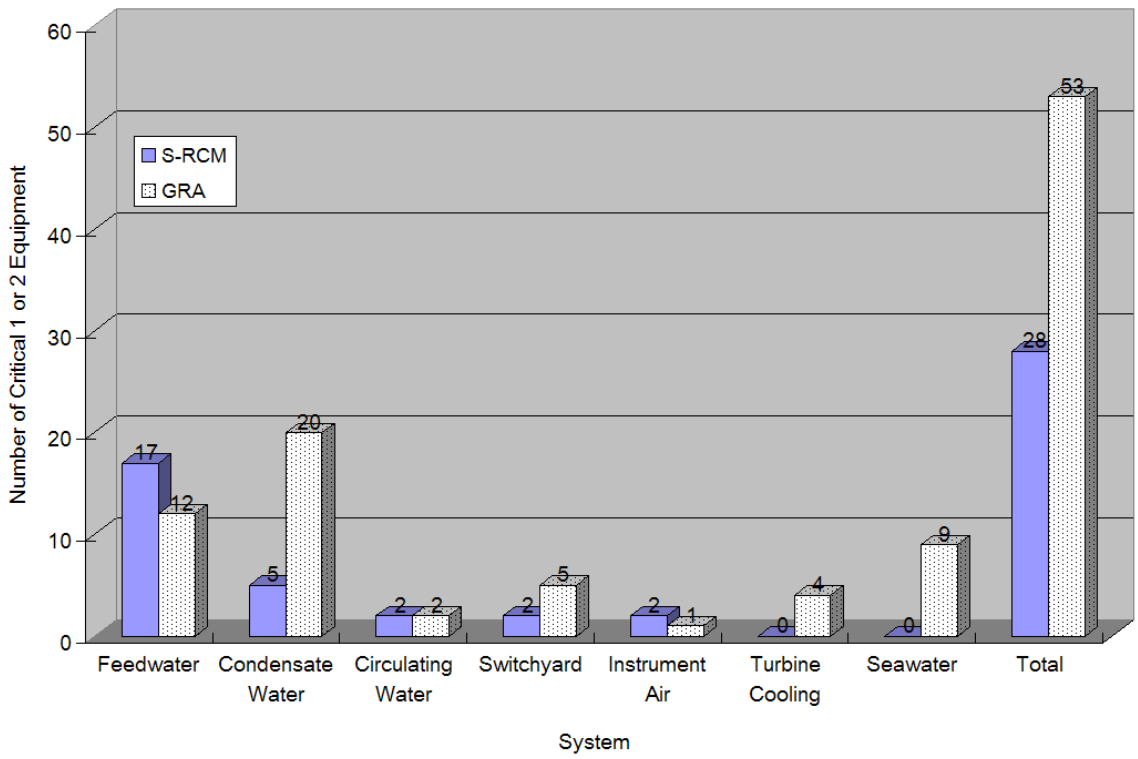

Figure 3-7
Distribution of equipment importance classification (Critical 1 and Critical 2) by system

The Condensate System results are directly opposed to those of the Feedwater System: GRA identified far more Critical 1 or 2 locations than did S-RCM (GRA identified 20 locations, whereas S-RCM identified only 5).

In the GRA model, support systems such as Turbine Cooling Water and Seawater were found to be of greater importance than was identified by the S-RCM process. If equipment in these systems would fail, the impact for the plant could be large because many systems could be simultaneously unavailable. The GRA model specifically addresses these interactions. In the S-RCM process, equipment importance of these support systems is lower because redundancy is high.

4
SUMMARY

Observations from this study include the following:

- Equipment importance classifications were similar using both GRA and S-RCM methodologies. A comparison of the two is illustrated in Figure 4-1. This methodology was used during the study, and it allowed for the selection of equipment importance based on a high-value selection process. Using this simplistic characterization could result in unnecessary maintenance tasks being considered and implemented on Non-Critical Systems, Structures, and Components. That fact accounts for some of the differences in equipment importance characterizations when compared to those identified when the methods were evaluated independent of the methodology (see Table 4-1).
- The GRA methods ranked 48 components of a population of 492 equipment locations (approximately 10%) higher in importance than did the S-RCM techniques. Table 4-1 shows the results of comparing the equipment importance characterizations between S-RCM and GRA.
- Single-failure mechanisms were detectable earlier using GRA methods. Evidence of this effect was more dominant in the Condensate System model; however, similar tendencies were seen in the Turbine Cooling Water and Seawater Systems.

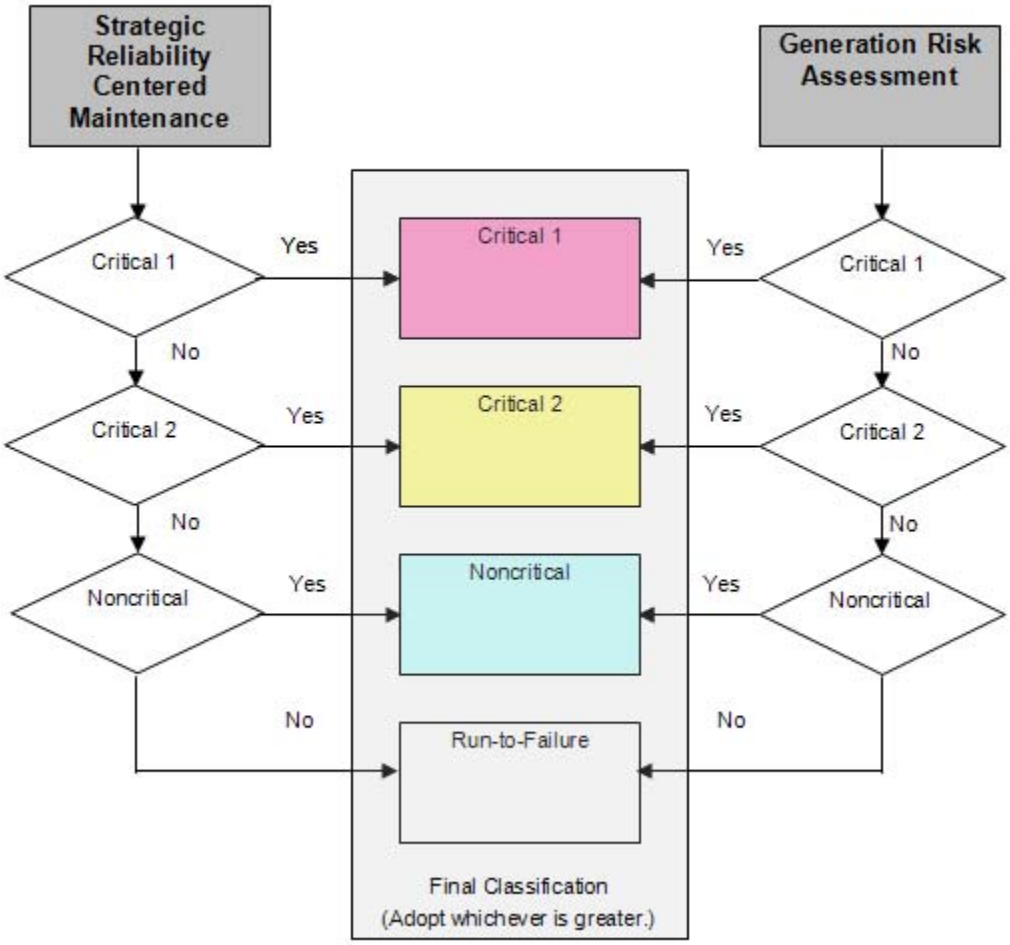

Figure 4-1
Flow diagram of equipment importance classification

Table 4-1
Equipment listing comparing classification by Strategic Reliability Centered Maintenance to Generation Risk Assessment methods

System	Equipment ID	Strategic Reliability Centered Maintenance	Generation Risk Assessment
Feedwater	MO-32-104A_M	Non-critical	2
Feedwater	MO-32-501_M	Non-critical	2
Feedwater	MO-32-503_M	Non-critical	2
Feedwater	FT-52-2A	Non-critical	1
Feedwater	32-C1A_P	Non-critical	2
Feedwater	FCV-52-3A_V	Non-critical	2
Feedwater	MO-32-104A_V	Non-critical	2
Feedwater	MO-32-501_V	Non-critical	2
Feedwater	MO-32-503_V	Non-critical	2
Feedwater	V-32-505_V	Non-critical	2
Condensate water	32-30-230_M	2	1
Condensate water	32-C4A_M	Non-critical	2
Condensate water	MO-32-7_M	Non-critical	2
Condensate water	LS-52-20	Non-critical	1
Condensate water	LSH-52-8	2	1
Condensate water	LSL-52-9	Non-critical	1
Condensate water	PSL-58-18A	Non-critical	2
Condensate water	PSLL-52-6A	Non-critical	2
Condensate water	FCV-52-1B_A	Non-critical	1
Condensate water	FIC-52-1B	Non-critical	1
Condensate water	FT-52-1B	Non-critical	1
Condensate water	32-10-107A_P	Non-critical	2
Condensate water	32-30-230_P	Non-critical	1
Condensate water	32-C3A_P	Non-critical	2
Condensate water	32-C4A_P	Non-critical	2
Condensate water	32-C4A_P	Non-critical	2
Condensate water	32-B1_C	Non-critical	1
Condensate water	30-30-B1	Non-critical	1
Condensate water	FCV-52-1B_V	Non-critical	1
Switch yard	ABB-O-3	2	1
Switch yard	275KV-PD-3L-R	Non-critical	1

Table 4-1 (continued)
Equipment listing comparing classification by Strategic Reliability Centered Maintenance to Generation Risk Assessment methods

System	Equipment ID	Strategic Reliability Centered Maintenance	Generation Risk Assessment
Switch yard	LS-3	Non-critical	1
Switch yard	275KV-CT-MTR-BP-R	Non-critical	1
Switch yard	275KV-CT-STR-R	Non-critical	1
Instrument air	V-36-101_V	Run to failure	1
Turbine water	36-C1A_M	Non-critical	2
Turbine water	36-C1A_P	Run to failure	2
Turbine water	V-36-3A-1_V	Run to failure	1
Turbine water	V-36-52_V	Run to failure	1
Seawater	MO-37-102A_M	Run to failure	2
Seawater	MO-37-5_M	Run to failure	2
Seawater	37-S-102	Non-critical	1
Seawater	36-010-001A	Non-critical	1
Seawater	MO-37-102A_V	Non-critical	2
Seawater	MO-37-5_V	Non-critical	2
Seawater	V-37-12A_V	Non-critical	1
Seawater	V-37-3A_V	Non-critical	1
Seawater	V-37-955_V	Non-critical	1

Cost-benefit analyses were not considered as part of this study; however, that would be a logical next step when evaluating assessment methodologies. Some TEPCO plants have completed S-RCM analysis using FMEA techniques for balance-of-plant components; most have not performed risk based analysis for the same equipment. Other areas to bear in mind would be the process of integrating S-RCM results with GRA results, the skill required to perform a risk based process versus an expert panel FMEA, and the time required to perform the analyses. Perhaps the value should be considered not only for existing plant programs but also for new plants, where generation risk based assessments would provide preventive maintenance programs at the time of plant delivery.

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Printed on recycled paper in the United States of America

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