GE Nuclear Energy

General Electric Company
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August 2, 1994

MFN No. 089-94
Docket No. STN 52-004

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

Attention: Richard W. Borchardt, Director
Standardization Project Directorate

Subject: NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design

References: Transmittal of Requests for Additional Information (RAI)s Regarding the SBWR Design, Letter from M. Malloy to P. W. Marriott dated April 8, 1994

The Reference letter requested additional information regarding the SBWR Isolation Condenser performance. In fulfillment of this request, GE is submitting Attachment 1 to this letter which transmits the response to RAI 440.6.

Sincerely,

Pinks Hasmit for

P. W. Marriott, Manager
Advanced Plant Technologies
M/C 781, (408) 925-6948

Attachment 1, "Response to NRC RAI"

cc: M. Malloy, Project Manager (w/2 copies of Attachment 1)
F. W. Hasselberg, Project Manager (w/1 copy of Attachment 1)

220098

RAI Number: 440.6

Question:

RAI SRXB.28 requested information concerning IC performance in currently operating BWRs and application of this information to assessment of IC performance for the SBWR. The staff requests that GE provide further information in this regard, to allow the staff to continue its review. Specifically, the staff requests that GE provide:

a. Operational data for ICs in current-generation plants, demonstrating the heat removal capability of these systems for extended periods of time. This includes heat transfer performance and accumulation (and possible venting) of non-condensible gases.

b. Assessment of the operational data demonstrating the applicability of the data to the IC design proposed for the SBWR, accounting for the effects of any design differences between the currently operating ICs and the SBWR system. This includes a quantitative comparison, using scaling methodology if appropriate (i.e., dimensionless parameters), showing that the range of thermal-hydraulic conditions experienced by currently operating ICs is similar to that predicted by GE for the SBWR. The range of conditions includes (but is not limited to) accidents during which the IC is expected to operate, regardless of whether specific credit for IC operation is taken in GE's standard safety analysis report analysis of accident.

c. Demonstration of the capability of the TRACG computer code to model the performance of the ICs in current plants, including the effects of non-condensible gases that can accumulate in the IC.

d. A detailed summary of the data from the COMPASS database from which GE drew reliability information presented in its revised response to SRXB.28 (MFN No. 103-93, June 30, 1993).

GE Response:

Item RAI 440.6 a.

There is a long and well-documented history of successful Isolation Condenser (IC) performance at operating plants. The design rules and practices for ICs are the same as for other heat exchangers used in licensed nuclear reactors such as the Reactor Heat Removal (RHR), Service Water, Feedwater Heaters, etc. Plant startup tests have confirmed that IC performance has consistently exceeded the conservative predictions of standard sizing methods as documented in this response. Operating plant experience has demonstrated that the ICs are a

simple and reliable way to remove decay heat following isolation. The methods used to size the SBWR IC heat exchangers are consistent with proven conservative methodology used to size operating plant ICs.

Table b.1 of this response contains a listing of plants with isolation condensers. The longest recorded use of an IC was at Dodewaard. Figure a.1 shows that after eleven (11) hours the heat removal curve appeared to follow the decay heat generation rate. There was no discernible reduction in performance due to buildup of non-condensibles. Dodewaard has eight hours of coolant inventory in the IC heat exchanger, the greatest of any operating plant. Therefore, it is expected that this is the longest continuous use of an IC in service.

No record of loss of IC heat exchanger efficiency due to buildup of non-condensibles has been found. The Oyster Creek FSAR 6.3 refers to a GPUN Topical Report No. 056 titled, "Evaluation of Isolation Condenser Performance with Non-condensible Gases in Steam". The FSAR states that: "The report concludes that closure of the vent line isolation valves or blockage of the vent line will not preclude proper operation of the system." Operational performance data for operating plants primarily consists of startup test results verifying the performance of the isolation condensers.

The following are excerpts from the startup tests found in GENE records. The general conclusion from the tests is that IC performance was up to 200% greater than predicted using standard heat exchanger sizing rules. This may be due to overly conservative fouling factors that are not appropriate for nuclear water quality systems. The excess capability of the ICs resulted in excessive water carryover at some plants requiring modification of the system to reduce flow. Water carryover has been addressed in the SBWR by the addition of a moisture separator at the discharge to the IC/PCCS pools.

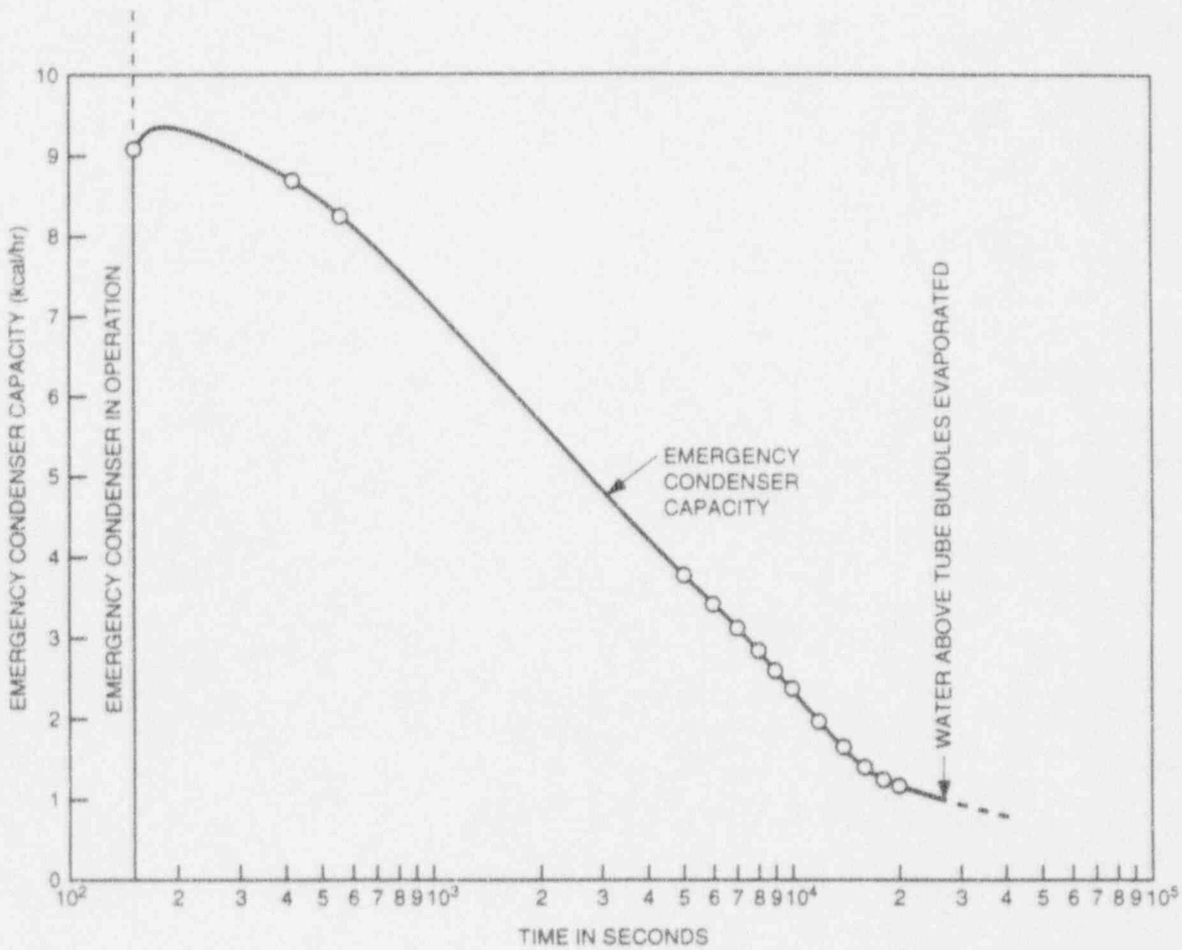

Figure A.1. Dodewaard IC Capacity

Isolation Condenser Startup Tests
Prior to plant operation each system must be tested to verify its performance conforms with design requirements. The following are extracts of the IC tests from the Dresden, Millstone, Nine Mile Point 1, Tarapur, Fukushima 2, Oyster Creek and Tsuruga startup test reports. Although each test was conducted using slightly different methods, all tests confirmed that IC performance exceeded design requirements by a significant margin.

A. DRESDEN II ISOLATION CONDENSER (Ref. NEDC-10430A)

Purpose
To determine the heat removal rate of the isolation condenser (IC) and the time for the system to come into operation. To demonstrate primary to secondary system leak tightness.

Criteria
The heat removal rate shall be no less than 250×10^6 Btu/h, and the system shall indicate primary to secondary leak tightness.

Results
The first Btu/hr heat removal capacity test was attempted at 10% reactor power with all steam bypassed initially to the main condenser. The capacity of the isolation condenser, and the decrease in reactor subcooling due to the isolation condenser return water, were sufficient to cause closure of the bypass valves and termination of the test.

The test was next attempted at the 25% power level but, because of inadequate indication of isolation condenser water level and excess carryover, the capacity was evaluated from a reactor heat balance instead of the shell-side heat balance as originally intended. Bypass valve movement was already calibrated as a function of feedwater flow thus enabling the following heat balance to be made.

As the isolation condenser was brought into service, the APRMs showed a decrease of 3% reactor power or 258.0×10^6 Btu/hr. This reduction in reactor power was due to a decrease in subcooling resulting from the IC return water. One bypass valve plus 35% of another bypass valve closed as the system went into service. The change in feedwater flow was determined to be 0.6×10^6 lb/hr. By assuming that the change in feedwater flow was equal to the change in steam flow to the main condenser, a heat balance was performed.

$$\Delta Q_{core} = \Delta W_{FW} \ (\Delta h_{core}) - \Delta Q_{IC}$$

$$\Delta Q_{IC} = -258 \times 10^6 \text{ Btu/hr} + (.6 \times 10^6 \text{ lb/hr})\ (1191 - 106 \text{ Btu/lb})$$

$$= 392 \times 10^6 \text{ Btu/hr or } 157\% \text{ Design.}$$

Automatic initiation of the isolation condenser was not employed for the test mentioned above because for operational reasons it was desirable to prevent the initial cold water slug associated with automatic starting of this system. However, the test was repeated later with automatic initiation, and a capacity

check was again made by the same method as for the earlier test. In this case the net bypass valve movement associated with establishment of the isolation condenser was 1.58 valves (SIC). The corresponding APRM decrease was again 3%, and substitution of these numbers in the heat balance equations gave a value of 480×10^6 Btu/h or 192% for the isolation condenser capacity. The reactor was initially operating at approximately 20% power with all steam being bypassed to the main condenser.

The isolation condenser was leak tight and the time to come into operation was measured to be 18 seconds. This time was defined as the time from the initial valve movement to the time that the bypass valves stopped moving. The readings of the gamma monitors on vents A and B increased from 0.1 mrem/hr to 1.5 mrem/hr when the isolation condenser came into service. This low value indicates that the system has no primary to secondary leakage.

During this test the steam and condensate return flow indicating switches were observed. In the case of the condensate return flow a maximum of 23" of water was recorded during operation of the condenser. The present setting of the condensate return flow trip is 32" of water.

Subsequently, on February 24, 1971, a design change was made wherein the opening of the condensate return valve was limited such that the effective capacity of the system was 120% of design. This modification was intended to limit the secondary carryove. and minimize the risk of system isolation from high steam flow on initiation.

With the reactor operating at 2330 MWt and 717 MWe with 1.65 bypass valves open and 96 percent core flow, the isolation condenser was brought into operation using condensate return valve 1301-3 to control flow (a special circuit was installed to interrupt the normal "seal-in" feature with a switch on the control panel by jogging the gate valve open or closed as desired). After two attempts, which resulted in calculated capacities of 132% and then 110%, the drive motor limit switch was placed at a position estimated to give about 120% capacity. Subsequent initiation of the isolation condenser resulted in a calculated capacity of 120% of design.

During the test of February 24, 1971 data was obtained to enable a calculation of carryover to be made. The calculation was based on a comparison of the actual isolation condenser water level drop rate, with the level drop rate equivalent to the condenser capacity with no carryover. The level drop rate based on 120% design capacity is calculated as follows:

$$
\begin{aligned}
Q_{IC}\ (\text{Btu/hr}) &= h_{fg}\ (\text{Btu/lb}) \times \rho\ (\text{lb/ft}^3) \times A\ (\text{ft}^2) \times R\ (\text{ft/hr}) \\
120\%\ \text{capacity} &= 300 \times 10^6\ \text{Btu/hr} \\
300 \times 10^6 &= 970 \times 543/0.01672 \times R\ (\text{ft/hr}) \\
R &= 9.55\ \text{ft/hr or } 0.16\ \text{ft/min (no carryover).}
\end{aligned}
$$

Using the measured water level drop rates, estimates were made of the carryover at different water levels for the shell-side of the isolation condenser. At a water level of about 6 feet, the carryover was calculated to be 8% by weight for water in the discharged mixture.

Because of general concern over the degree of carryover it was decided to modify the isolation condenser by fitting baffles to reduce the water loss. Following these modifications further carryover tests were made on May 27, 1971. The table below shows the change of shell-side water level with time.

ISOLATION CONDENSER CARRYOVER

Indicated Shell-Side Level (ft)	Level Drop Rate (ft/min)	Less Expected Rate (ft/min)	Carryover lb Water/lb Mixture
6.3	0.64	0.48	75
5.2	0.39	0.23	59
4.4	0.27	0.11	41
4.0	0.22	0.06	27
3.6	0.22	0.06	27
3.1	0.22	0.06	27
2.5	0.18	0.02	11
2.2	~0.16	~0.00	~0

The test of May 27, 1971 also indicated an effective capacity of 120 percent of design, thus again confirming the throttled setting of the condensate return valve; the calculation of capacity being made as for the previous tests. The isolation condenser system was again demonstrated to be leaktight, as evidenced by the relatively small increase in the "A" and "B" vent radiation monitors from 0.2 and 0.4 mrem/hr before the test, to 3.1 and 3.0 mrem/hr, respectively, during the test.

Discussion

The capacity of the isolation condenser was shown to be more than adequate. Primary to secondary leak tightness was demonstrated and a satisfactory automatic initiation time recorded.

The purpose of the isolation condenser may be expressed as follows:
To provide 16.3 minutes of operation at 310 MBtu/hr without adding makeup to the shell-side.

For the following reasons the May 27, 1971 test was considered to be successful on this basis:

a. The test duration was 16.5 minutes.
b. The shell-side water temperature was initially 212°F rather than the normal 90-100°F.

c. The pre-warming of the shell-side water resulted in the loss of about 0.4-0.5 feet, so that the initial shell-side inventory was less than normal.

d. The test could have been extended for some time longer. There was no particular reason for stopping it at 16.5 minutes.

Carryover, as measured during the May 27, 1971 test, was shown to be strongly dependent upon the shell-side water level. However, the test demonstrated that an adequate quantity of water existed without makeup and that carryover was in fact almost zero below a shell-side water level of 2.5 feet.

B. MILLSTONE 1 ISOLATION CONDENSER TEST (STP 13)
(Ref. NEDE-13201A)

Purpose
Determine the heat removal rate of the system. Determine the time for the system to come into operation. Determine that the system is leak tight.

Criteria
Determination of the heat removal rate of the isolation condenser, determination of the time for the system to come into operation, and verification of system leak tightness will constitute satisfactory completion of the test.

The expected heat removal of the isolation condenser is about 206 million Btu/hr.

The time for the system to come into operation is expected to be on the order of one to three minutes.

The isolation condenser is expected to be leak tight under all conditions.

Results
The valve on the steam line is normally open so that the isolation condenser was put into service by controlled opening of the condensate return valve. The automatic shell side fill system was inhibited by closure of a manual valve and the stationing of an operator at that location to put the automatic fill system in service if required.

The time to put the system in service was based on analysis of the transient recorder trace from first movement of the condensate return valve to the time bypass valve cam position reached steady conditions.

The capacity of the system was determined from steam flow change through bypass valves and a condensate return flow at 370°F. Leakage of reactor water through the heat exchanger tubes could be detected by the radiation monitor on the vent to atmosphere. Test result is shown in Table C.8-1.

The isolation condenser was put into service in 26 seconds and has a capacity of 2.5 times rated.

The radiation level in the condenser increased 0.1 mr/hr, indicating satisfactory leak tightness.

On fast activation, there was a high flow problem in the differential pressure detectors in the line leaving the reactor vessel and in the return line. Therefore, the isolation condenser return line valve was set to open to prevent high flow and subsequent isolation. After the valve was adjusted, the isolation condenser was tested again and its capacity determined by a tube side heat balance. This test was performed by fast actuation of the return valve and the system came into service in 3 to 5 seconds.

The capacity was determined to be 373 MBtu/hr or 5.4% of rated power. This is a factor of 1.8 times rated capacity for the system.

Table C.8-1
ISOLATION CONDENSER TEST RESULTS

Test Conditions

Reactor Power	25%
Core Flow	~35%
Bypass Cam Position	25%
Bypass Valve No. 1	full open
No. 2	93%
Turbine Generator	Off Line

		Before Test	IC in Operation	End of Test
Bypass Valve No. 2 % Open		93	20	—
IC Vent Radiation	Monitor 1	0.2	0.3	0.8
Level (mr/hr)	Monitor 2	0.1	0.2	0.6

Q_{IC} = (Bypass Steam Flow Change) x (hg - h_{return})

= 0.73 x 833,000 lb/hr x (1199.3 - 344.1) Btu/lb

= 520 MBTu/hr, 7.6% of rated power

Where, Q_{IC} = heat removal rate of isolation condenser

hg = main steam enthalpy (reactor pressure = 979 psig)

h_{return} = condensate return line enthalpy (return line temperature = 370°F).

C. NINE MILE PT. 1 ISOLATION CONDENSER TEST (Ref. NEDE-10278)

<u>Purpose</u>
To determine the heat removal rate of the system, the time for the system to come into operation, and demonstrate the leaktightness of the system.

Criteria
Determination of the heat removal rate of each set of condensers, response time of the system, and verification of system leaktightness constitute satisfactory completion of the test.

Results
The isolation condenser system is comprised of four individual condensers. The shell side of each condenser is water-filled and is vented to the atmosphere. The tube side (the flow path through which vessel steam flows) is condensed and then returned to the recirculation loop through a condensate return valve. The condensers are paired such that each of two condenser sets shares common steam supply and condensate return piping.

Each isolation condenser set was tested individually. The reactor core thermal power before initiating condenser operation in each instance was approximately 200 MWt. Data were recorded both from process instrumentation on the control room panels and on the transient recorder. Condenser capacity was calculated from the control room instrumentation data using both a shell-side and a tube-side heat balance. Capacity was also calculated using a tube-side heat balance based on the transient recorder data.

The calculated capacities are given in C-3 below, "ISOLATION CONDENSER Calculated Capacity". Due to meter fluctuations, the transient recorder produced the most precise data. Table C-3 also presents the capacities calculated using these different data sources. The most drastic of the fluctuations mentioned above was noted on the shell-side water level instrumentations. The fluctuations ranged almost over the entire scale of the instrument. The source of these fluctuations was obviously the varying shell-side water density as a result of the boiling. The subjectivity used in reading the shell-side water level is reflected in the capacity calculated with these data. Likewise, the lesser magnitude fluctuations in the process instrumentation introduced subjective error in the tube-side heat balance. The permanent record obtained from the transient signal recorder made possible a more precise evaluation of the plant parameters.

Table C-3
ISOLATION CONDENSER Calculated Capacity

Condenser Set Number	Calculated Capacity (10^6 Btu/hr) A	B	C	Design Minimum
111/112	996	520.47	405.64	190
121/122	453	223.78	370.86	190

A = Shell side heat balance from process instrumentation
B = Tube-side heat balance from process instrumentation
C = Tube-side heat balance from transient recorder data

Assuming an absolute error in the transient recorder data of +/-2.5% (based on bypass valve position oscillation), the minimum capacity of the condensers would be: (a) 222.14 (10^6) Btu/hr for condenser set 111/112 and (b) 187.36 (10^6) Btu/hr for condenser set 121/122.

The elapsed time required for the system to come into operation is defined by the greater of: (a) the condensate return valve opening time, or (b)the time required for the main steam bypass valves to reach their new position. The time required for the condenser to come into operation was: (a) 3 minutes for condenser set 111/112 and (b) 1.5 minutes for condenser set 121/122.

The leaktightness of the system was measured by the change in radiation level at the isolation condenser shell-side atmospheric vent. The measured change is shown in Table C-4. The system was considered leaktight since the change in magnitude was extremely small. This observed change in radiation level was attributed to the increase in steam flow through the condenser near the radiation monitors.

Table C-4
ISOLATION CONDENSER VENT RADIATION DATA

Condenser Set	Radiation Level (mR/hr) Before Operation	During Operation
111/112	0.340	0.385
121/122	0.234	0.310

D. TARAPUR EMERGENCY CONDENSER (Ref. NEDE-13111)

Emergency condenser capacity can be measured by two methods: a shell side heat balance or a tube side (reactor) heat balance. The shell side heat balance is made by measuring the evaporation rate of the condenser cooling water. The rate at which the condenser coolant evaporates is measured by the rate-of-change of water level in the condenser tank.

By shell-side heat balance, the heat removed by the emergency condenser is:

$$Q_{ec} = W \times (h_s - h_f)$$

where

h_s = enthalpy of saturated steam at atmospheric pressure

h_f = enthalpy of saturated water at atmospheric pressure

W = rate of evaporation = $[\Delta L \times A)/t] \times \rho$

where ΔL = change in condenser water level in time 't'

A = condenser area of cross section

and ρ = density of water near 100°C

The tube side heat balance is given by the equation:

$$Q_{ec} = (W_{ps1} - W_{ps2})h_s - (Q_{c1} - Q_{c2}) - (W_{fw1} - W_{fw2})\, h_{fw}$$

where subscript 1 refers to measurements made just before the emergency condenser operation and subscript 2 refers to measurements made during the emergency condenser operation.

W_{ps} and W_{fw} = primary steam and feedwater flow readings
Q_c = core thermal power, given by the average of the six Power Range Monitor (PRM) readings
h_s and h_{fw} = enthalpies of saturated steam and water at reactor pressure.

Recirculation pump power addition, heat lost by cleanup, and rod drive cooling systems are assumed constant under both conditions. Negligible secondary steam was being used for gland seals and air ejectors during this test.
Unit 1 was the first emergency condenser to be tested. During the initial power operation of Unit 1 (Phase IV) the emergency condenser was put into operation. This test operation was with one tube-bundle at a time (of the two in the unit) in order to avoid reactor depressurization, which could be possible if the heat rejection through the emergency condenser was greater than the reactor power. Once the single tube-bundle capacity was established, the condenser was again tested with both tube bundles in service, and the reactor at a power level high enough to avoid depressurization. With the experience of the Unit 1 testing, Unit 2 was tested only once, with both tube bundles in service. The results obtained from the testing are given in Table 14A-1.

Table 14A-1
EMERGENCY CONDENSER CAPACITY

	Tube Bundle 1 Only	Tube Bundle 2 Only	Sum of Tube Bundles 1 & 2	Both Bundles In Service	Unit 2 Both Bundles In Service	Design Value Both Bundles In Service
From Tube Side Heat Balance	19.9 MW	21.2 MW	41.1 MW	67.6 MW	46.5 MW	
From Shell Side Heat Balance	31.7 MW	31.2 MW	62.9 MW	69.4 MW	59.2 MW	39.6 MW
Core Power Before Testing		50 MW		302 MW	132 MW	

Results from the shell side heat balance calculations show reasonable agreement between the two units. The tube side heat balances performed at low power levels are grossly inaccurate mainly due to the error in the measurement in primary feedwater flow at low flows. The accuracy of the shell side heat balance calculation is independent of the core power level, and is reasonable at all power levels. Good agreement was found among all calculations made by this method.

During all testing, the emergency condenser's vent side gamma activity was monitored and found to be less than 0.1 millirem/hour, a negligible quantity. A visual inspection of the condensers during operation indicated no serious interference between structures due to thermal conditions, and no vibration problems.

E. FUKUSHIMA 2 ISOLATION CONDENSER (Ref. NEDE-10426)

Purpose
To determine the heat removal rate of the isolation condensers and the time for the system to come into operation, and to demonstrate that the system is leak tight.

Criteria
Level 2
The expected heat removal rate of each isolation condenser is about 36.2×10^6 Kcal/hr.

The time for the system to come into operation is expected to be on the order of one to three minutes.

The isolation condensers are expected to be leak tight under all conditions.

Results
The isolation condenser is comprised of two individual condensers. The shell-side of each condenser is water-filled and is vented to the atmosphere. The tube-side (the flow path through which vessel steam flows) is condensed and then returned to the recirculation loop through a condensate return valve. The condensers are paired such that each of the condensers has separate steam supply and condensate return piping.

Each isolation condenser set was tested individually. Data were recorded both on process instrumentation in the control room and on the transient recorder. Condenser capacity was calculated from the control room instrumentation data using both a shell-side and a tube-side heat balance. Capacity was also calculated using a tube-side heat balance based on the transient recorder data.

The reactor was operating at 234 MWt (17% rated) with all the steam bypassed to the main condenser. The feedwater system was in manual mode, and the MPR (Mechanical Pressure Regulator) was in service. Makeup water to the isolation condensers was valved out of service.

Table C.6.1 summarizes the results of the measurements and shows that the measured capacity was much greater than designed. The disparity between shell and tube calculations would indicate a large amount of liquid carryover from the condenser shell. This, in fact, was observed at the condenser vent during the test. The carryover was large enough that it was decided to reduce the steam flow to the condenser by limiting the opening of the valve in the return line.

The isolation condenser was tested again at 50% power to find the return line valve opening that best matched the measured and design heat removal capacity of the isolation condenser. The results are presented in Figure C.6.1. It was decided to limit valve V-1301 to about 15% opening. However, this was never implemented because of the complexity of the valve's positioning mechanisms.

Additionally, radiation levels at the shell-side vent locations were monitored during the tests. Only negligible increases in the radiation levels were observed. Likewise steamline pressure differences were monitored. These are shown in Table C.6.2.

Table C.6.1
ISOLATION CONDENSER CAPACITY

Isolation Condenser	Capacity (10^6 Kcal/hr) Shell-Side Measured	Tube-Side Measured	Design	Time to Operation (min)
A	68	48	36	1.8
B	123	54	36	1.8

Table C.6.2
SHELL-SIDE VENT RADIATION LEVELS

Isolation Condenser In Service	Before/During	Radiation Sensor Reading (mr/hr) A[a]	B[a]	C[a]	D[a]
A	Before	0.2	0.3	0.2	0.2
	During	0.4	0.6	0.5	0.5
B	Before	0.2	0.3	0.21	0.2
	During	0.23	0.41	0.31	0.31

(a) Radiation Sensors A/B Monitor Vent A; Sensors C/D Monitor Vent B

Shell and Tube-Side Temperatures and Differential Pressures

Isolation Condenser In Service	Before/During	Temperature (°C) Shell	Tube A/B[c]	C/D[c]	Differential Pressures (%)[b] Steam A/B[c]	C/D[c]	Water A/B[c]	C/D[c]
A	Before	46	54	55	0	0	0	0
	During	88	168	165	38	40	17	16
B	Before	43	42	41	0	0	0	0
	During	88	163	175	40	40.5	18	18

(b) Differential pressures are monitored in tube-side piping for pipe breaks.
Steam - 635 cm H_2O = 100%
Water - 508 cm H_2O = 100%

(c) Sensors A/C monitor condenser A; Sensors B/D monitor Condenser B.

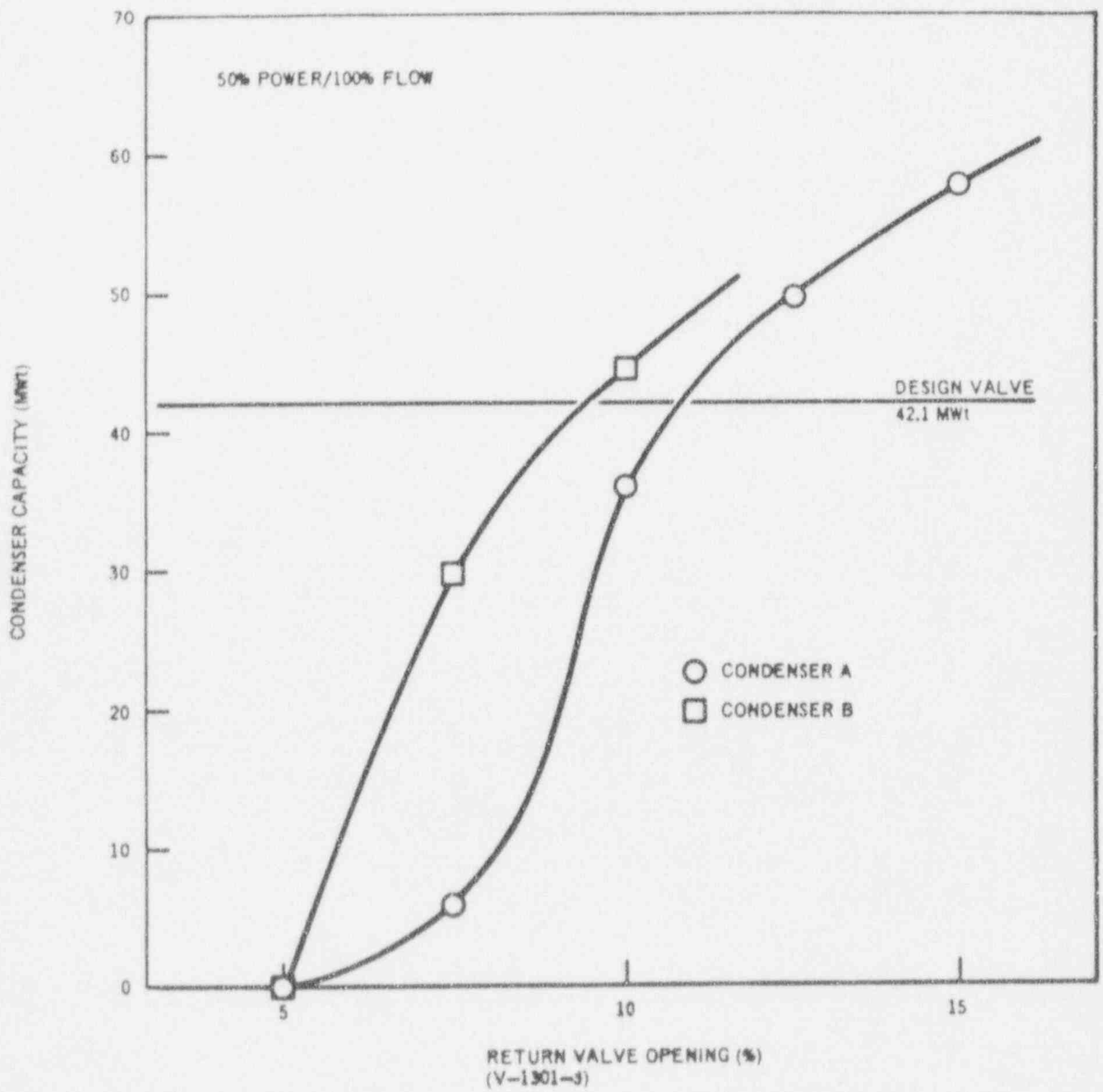

Figure C.6.1 Isolation Condenser Capacity Versus Valve Opening

F. OYSTER CREEK ISOLATION CONDENSER TEST (Ref. NEDE-13109)

Purpose
To determine the heat removal rate of the isolation condensers and the response time of the system.

Results
The test was performed at the following reactor conditions

Reactor Power	200 MWt
Reactor Pressure	990 psig
Core Recirculation Flow	58×10^6 lb/hr
Vessel Steam Flow	470,000 lb/hr

The condensate return line valve was tripped to initiate the transient. Condenser water level was recorded using plant instrumentation and the transient was recorded on the Sanborn recorder. Heat removal rates were calculated for both condensers using shell-side and tube-side heat balances.

Tube-side heat balance

$$Q = (W_{s1} - W_{s2})\,(h^c_g - h^c_f)$$

where:

W_{s1}, W_{s2} = Turbine steam flow before and after condenser initiation
m_{g2} = main steam flow after isolation condenser initiation
h^c_g = enthalpy of steam entering the condenser
h^c_f = enthalpy of fluid leaving the condenser

Shell-side heat balance

$$Q = \rho\ \Delta V/\Delta T\ h_{fg}$$

ρ = density of fluid in condenser
$\Delta V/\Delta T$ = Change in volume of water with time
$h_{fg} = h_g - h_f$ Evaporation enthalpy at shell temperature

The measured heat removal rates for each condenser are given in Table D.3.1. Both condensers have a measured capacity of at least 70 MWt, compared to a design capacity of 56 MWt. The response time of the system, defined as the time for the bypass valves to reach steady state, is about 10 seconds. Figure D.3.1 shows the transient traces obtained on the Sanborn recorder for the test of isolation Condenser A. Power (neutron flux) rose about 20% above its initial value and settled out slightly below the initial condition. There were two effects which caused the power swing. First was the leg of cold water entering the reactor when the condenser return line was opened, and second was the effective increase in feed-flow from the condenser since the feedwater system was on manual control and, therefore, did not respond to level and steam flow

changes except by operator action. Pressure initially dipped 10 psi and returned to its original value within 15 seconds. Water level rose 7 inches and slowly returned to its initial value. Radiation levels measured at the condenser were not excessive, indicating satisfactory leak tightness. The increases in radiation level that were seen near the condensers are attributed to the flow of reactor steam through the system.

Each condenser blew significant amounts of water out of the vents during the first 5 or 6 minutes of shell-side boiling, therefore, a heat balance based on change in shell-side water level used the rate of level change after this period of excessive carryover settled out.

Table D.3.1
ISOLATION CONDENSER HEAT REMOVAL

	Condenser A (MWt)	Condenser B (MWt)
Shell-side heat balance	131	86.5
Tube-side heat balance	70	82.5

G. TSURUGA ISOLATION CONDENSER (Ref. NEDE-10224)
November 23, 1969 to November 23, 1969

Purpose
To determine the heat removal rate of each condenser and the response time for the system, and to demonstrate the leak tightness of the system.

Criteria
Level 1 Determination of the heat removal rate of each isolation condenser, determination of the time for the system to come into operation, and verification of system leak tightness constitute satisfactory completion of this test.

Reactor Conditions
Reactor thermal power 168 MW, reactor pressure 67.8 kg/cm^2, bypass valve 70% open, isolation condenser temperature 98°C.

Results
The response time and heat removal rates calculated from the tests of the isolation condensers are shown below.

	Condenser A	Condenser B
Response time (sec)	10	14
Shell side heat balance (MWt)	65	46.5
Tube side heat balance (MWt)	62.2	60.8

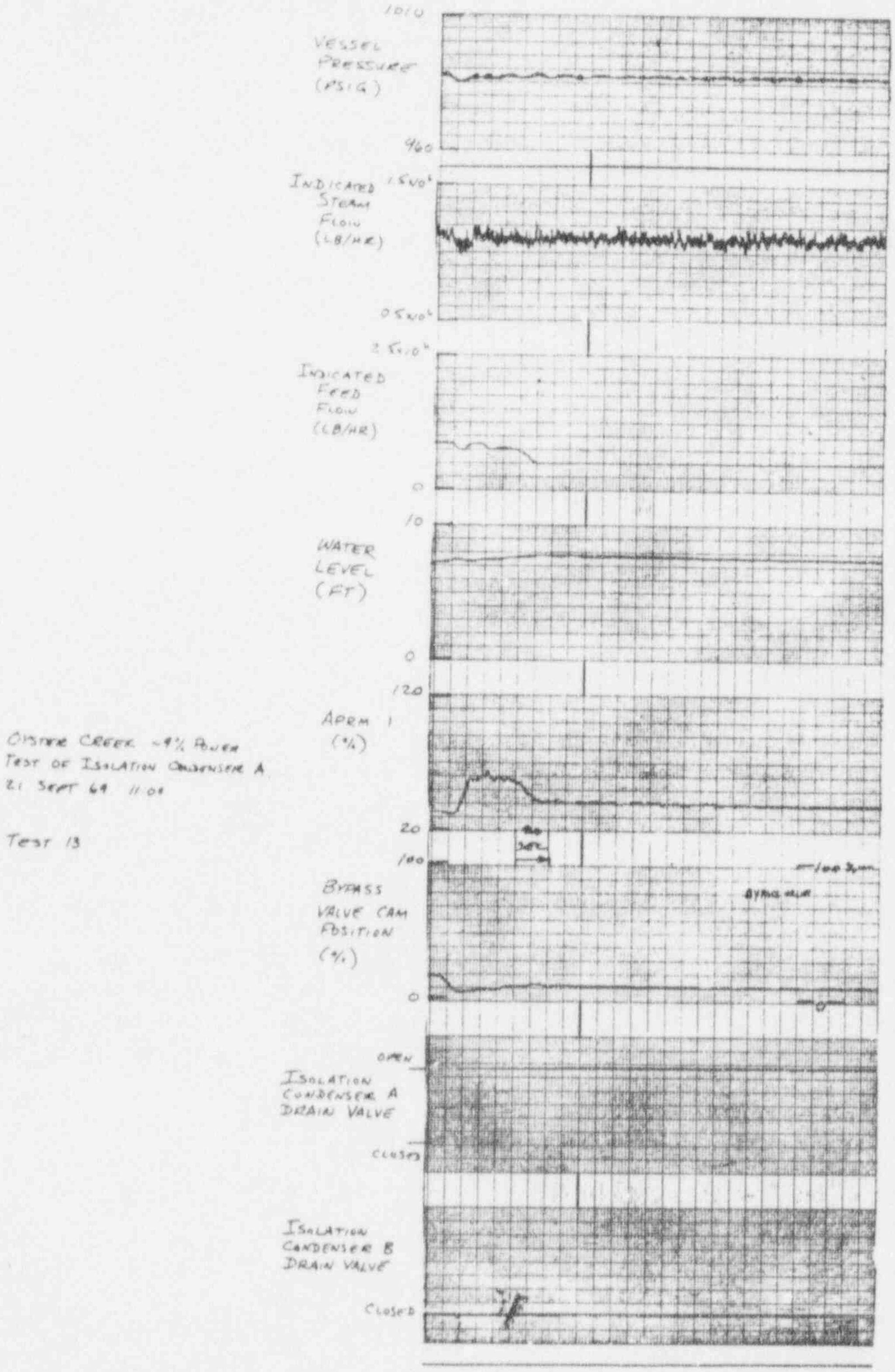

Figure D.3.1 Isolation Condenser Test, Run 6

Discussion

The tests of the isolation condensers were performed with the bypass valve initially open about 70% and with the condensers isolated from the water storage tanks. The bypass valve position was recorded during the test and, after the condenser return valves were closed, the volume of water drained from the storage tanks to fill the condenser shell back to its original level was determined. The heat removal rate calculated from the change in bypass valve position is considered to be a more reliable measure of steady-state condenser capacity than the shell side heat balance because there are fewer variables involved in the calculation and the determination of changes in the parameters of interest are not as prone to the influence of outside variables such as carryover.

Using the design bypass valve capacity (16.7% of rated power of 968.4 MWt) the capacity of each of the isolation condensers is at least 60 MWt or about 200% of the design capacity of 31.9 MWt per condenser. It was visually verified during operation that the carryover at the shell side vents was insignificant.

When the condensers were put in operation the neutron flux went through a one-cycle oscillation of about 10% and then settled out at essentially the initial value. The bypass valve closed to an equilibrium value of about 30%. The feedwater system was being operated manually during the test to maintain reactor water level, and neutron flux, and bypass valve opening increased in phase with the feedwater injection. However, these traces consistently returned to the same deflection between feedwater transients. The reactor pressure dropped very slightly (~ 0.125 kg/cm^2) while the bypass valve was closing during the initial transient but quickly returned to and maintained the original pressure during the transient. The radiation level in the condensers increased less than 0.1 mr/hr, indicating satisfactory leak tightness.

Item RAI 440.6b.

Table b.1 is a tabulation of operating plant IC parameters compared to the SBWR. Most operating plants and the SBWR ICs have been sized to handle approximately 3% reactor thermal rating. The major difference between most operating plants and the SBWR is that the SBWR has three (3) ICs of 1-1/2% capacity where most operating plants have one or two condensers of 3% capacity. Also, the SBWR has an IC coolant inventory sized for 72 hrs compared to between eight hours and one-half hour for the operating plants. Operating plant IC heat exchanger thermodynamic parameters used to calculate heat removal capability are based on tube side saturated steam pressure of approximately 1,000 psia and shell side temperatures of approximately 228°F. The SBWR IC condenser capacity is based on 1,050 psia saturated steam on the tube side and shell side temperature of 212°F. The SBWR IC is in an open pool, rather than a slightly pressurized vessel as in operating plants, which accounts for the lower shell side temperature. The SBWR IC has vertical tubes, whereas the operating plant units have horizontal tubes. The range of operating plant IC thermal-hydraulic conditions is similar to that expected to be experienced by the SBWR. Therefore, operating plant IC experience is clearly applicable to the SBWR.

Item RAI 440.6 c.

Capability of the TRACG computer code to model the performance of the ICs, including the effects of non-condensible gases that can accumulate, will be evaluated and documented following completion of the SBWR Testing and Analysis Program (TAP).

Table b.1
OPERATING PLANT VS SBWR
ISOLATION CONDENSER PARAMETERS

	Reactor Thermal Megawatt	Number of Condensers	Bundles per Shell	Capacity hr. and (lb. water)	Minimum IC Capacity per IC % Thermal Megawatt
Dresden I	700	1	2	Note 4	Note 4
Humbolt Bay	165	1	Note 4	Note 4	Note 4
Garigliano	508	1	2	Note 4	24
KRB-A	800	1	2	4	6
Big Rock Point	157	1	2	Note 4	
Tarapur 1 & 2	1,322	1	2	4(3) (183,000)(1)	3
Dodewaard	165	1	2	8(3)	6
Nine Mile Pt. 1	1,538	2 sets of 2	1	1.5(2) (170,000)(1)	3.6 per set
Oyster Creek 1	1,860	2	2	1.5(3) (170,000)(1)	3
Dresden II, III	2,527	1	2	Note 4	3
Millstone 1	2,011	1	2	0.5(3) (129,000)	3
Tsuruga	1,064	2	2	Note 4 (117,200)(1)	3
Nuclenor	1,381	1	2	1(3) (105,000)(1)	3
Fukushima 1	1,380	2	2	Note 4	3
OKG 1	1,500	1	2	8 (500,000)	6
SBWR	2,000	3	2	72 (3,083,700)	1.5

Notes:

1. The amount of water above the IC condenser tubes is the minimum specified in the IC condenser purchase specification and may be significantly greater in the as-built heat exchanger.

2. An additional eight hours of cooling water is available from makeup tanks.

3. The time of available cooling capability is based on decay heat rate curves and the amount of cooling water available above the IC tube bundles. Times will vary depending on decay heat assumptions and as-built tank capacity.

4. Values not found in IC purchase specifications or specification not found in GENE records.

Item RAI 440.6 d.
Provided below are summary reports from COMPASS and the Nuclear Plant Reliability Data System (NPRDS) on the reliability of ICs in service.

Summary Report on NPRDS Data on IC Failures in Operating BWRs

NPRDS (Nuclear Plant Reliability Data System) is operated by the Institute for Nuclear Power Operations (INPO) to collect component failure rate data from operating nuclear plants. On September 26, 1990, NPRDS was queried for data on all failures in isolation condenser systems at U.S. BWRs. Failures were reported at the following operating plants:

Dresden II
Dresden III
Oyster Creek
Nine Mile Point 1
Millstone

NPRDS responded with data dating back to 1974 (the approximate time that plants started reporting data to NPRDS). Using a factor of 70% average plant availability, an estimate of the total plant operating time for this span of data is 450,000 hours over 50 reactor years for the above plants, which represent only about half of the IC equipped BWR fleet.

The components included in the NPRDS definition of the IC system, and the number of failures reported for each component are as follows:

Accumulator	0
Annunciator	0
Circuit Breaker	0
Heat Exchanger	2
Bistable Switch	49
Level Controller	2
Indicator/Recorder	5
Integrator	6
Power Supply	0
Transmitter	15
Pipe	29
Relay	14
Support	33
Valve	31
Valve Operator	36

Summary discussions of the reported failures for each of the components follow:

- Heat Exchanger: Only one of the two reported heat exchanger failures contained narrative description, and that narrative was unclear except to indicate that the heat exchanger was declared inoperative due to "line break sensing". (Note: The heat exchanger failures are believed to be for Millstone 1 due to a salt water intrusion into the reactor.)

- Bistable Switch: Of the 49 reported failures, 43 were reported to be due to instrument setpoint drifting out of technical specification limits in the high-flow delta-pressure sensor. This recurring problem occurred at all plants except Nine Mile Point and continues to the present time at most plants. The remaining 6 failures were random.

- Transmitter: Of the 15 reported failures, 9 were due to the same setpoint drift problem as above, except that at Dresden III they were classified as transmitter failures. The remaining 6 were due to miscellaneous random problems.

- Level Controller: The two level controller failures were random and unrelated.

- Indicator/Recorder: The five reported failures were random and mostly due to wear and aging.

- Integrator: The six reported failures were random and largely due to aging.

- Pipe: Of the 29 reported pipe failures, only 15 contained descriptive narratives. These 15 failures were all due to cracks in lines external to the IC, mostly outside of the drywell.

- Relay: The 14 reported relay failures were due to various causes and apparently did not affect plant operation.

- Support: All but one of the failures were due to pipe snubber problems.

- Valve/Valve Operator: Although the data classifies 31 failures as valve failures and 36 as valve operator failures, the classifications are not always distinct. Of the 67 total reported failures, 13 were failures of valves to open and 12 were failures to close. In addition to the 12 failures to close, there were an additional 14 reported cases of excessive leak-through. It is sometimes difficult to distinguish between a leak-through and a failure to completely close. Additionally, there were 7 reported failures of excessive out-leakage. The remaining 21 failures were miscellaneous problems involving valve packing, torque switches, limit switches, and other items.

Of the total 67 valves, 32 were either unidentified or the identity of the part number was not known. Of those that were identified, 15 were in the condensate return line. An unknown number of these were isolation valves. There were 11 reported failures in steam inlet isolation valves, 7 failures in vent valves, and 2 failures in level control valves.

From the information included in the report it was not possible to determine the effect of individual failures on plant operation. However, the availability of IC appears to be high compared to other more complex systems. The vast majority of problems have been with electrical components. The isolation condenser itself is highly reliable and would be operable even in the case of leaking tubes.

Summary Report on COMPASS Data on IC Failures in Operating BWRs

COMPASS (Comprehensive Performance Analysis and Statistics System) is the GENE reliability data system for operating plants. This system receives daily reports from most domestic operating plants. Data on equipment performance is mostly in relation to failures that result in plant outages or power reductions. Also included are critical path and other important activities that are conducted during plant outages. Not included are routine maintenance activities conducted while the plant is operating. Also not included are descriptions or logging of test activities or failures found during test (unless the failure resulted in plant shutdown).

On October 2, 1990, COMPASS was queried for data on all failures in isolation condenser systems at the following operating plants:

Dresden II
Dresden III
Oyster Creek
Nine Mile Point 1
Millstone

COMPASS responded with data dating back to 1970 and into 1989. The total calendar time for these 5 plants is 874,000 hours. The total operating time for these plants is 607,000 hours or over 70 reactor years of operation. This represents about half of the IC equipped BWR fleet.

The IC components that had reported failures and the number of failures for each component are as follows:

Cable	3
Heat Exchanger	5
Instrumentation	9
Pipe/Nozzle/Safe End	19
Support	5
System	6
Valve	57

Summary discussions of the reported failures for each of the components follow:

- Cable: The three cable entries were to replace cables with environmentally qualified cables at NRC request.

- Heat Exchanger: One of the heat exchanger failures was for a major modification; one was for sand-blasting and painting; and one was for repair of a crack in a diaphragm weld. The other two failures were tubing leaks, in one case requiring a complete retubing.(Note: Believed to be Millstone 1 after a salt water intrusion.)

- Instrumentation: Two of the instrumentation failures involved a timer, three were due to setpoint drift, and four were miscellaneous or unspecified failures.

- Pipe/Nozzle/Safe End: Sixteen of the piping failures were due to cracking, primarily due to IGSCC. Two of the failures were due to water hammer problems, and one failure was to replace a head gasket on a steamtrap.(Note: Two water hammer incidents occurred due to condensate pockets in the steam lines caused by poor drainage. Addition of drain lines eliminated the pockets and water hammers have not reoccurred.)

- Support: All of the support entries were for problems with snubbers.

- System: Three of the system entries were for conducting heat capacity tests, and two entries were for miscellaneous problems. Two of the entries involved contaminated spills outside of the plant.

- Valve: The distribution of the valve failures according to type of valve is as follows:

check valves	3
vent valves	7
isolation valves	22
solenoid valves	1
condensate return valve	3
general (unidentified)	21

There is no indication in the data as to how many of the unidentified valve failures might be failures in the condensate return valves.

The distribution of the valve failures according to type of failure is as follows:

packing leaks	13
limit or torque switches	7
broken/bent stems	2
misc./unspecified	35

Of the three identified condensate return valve failures, the nature of two of the failures was unspecified. The other was a failure of the valve to open during a pressure transient resulting in operation of a safety relief valve. This failure to open was due to either a faulty contact or a torque switch being out of adjustment.

Of the 104 total events recorded, 15 caused the reactor to be shut down including one automatic and one manual scram. In 37 of the events, IC work was on critical path contributing approximately 4216 hours to plant outage time, approximately 0.5% unavailability.

The apparent largest contributor to the reported event frequency was the valve, causing 57 of the 104 entries. Valves also caused 11 of the 15 reported IC causes of plant shutdown, including the single automatic scram. In contrast, valves contributed only about one-fifth (881 hours) of the total critical path unavailability, a proportionately small amount. From the reported data, it is not possible to determine how many of the valve failures would have prevented proper system operation on demand, but it would appear to be only a small fraction of the entries.

Another item of interest is the heat exchanger. There were only 5 heat exchanger events reported, and none of them was a failure that would have prevented the IC from responding to a demand. In addition, the heat exchanger events contributed only 43 hours of critical path unavailability. However, the total component (and system) outage time for maintenance was 1565 hours, and one of the events caused the plant to be operated at a reduced power level of 25-40% for about 5 weeks while tube repairs were being made. Most heat exchanger repairs were made off of the critical path during outages for other work.

The largest contributor to critical path plant unavailability was piping (3125 of the total IC contribution of 4216 hours). Piping also required 11,220 hours of component and system unavailability due to maintenance, repair, or replacement. Almost all of this unavailability was due to cracking, probably mostly due to IGSCC.

Setpoint drift in the IC differential pressure instrumentation, which has been a chronic problem, did not appear as a major contributor. This is because most setpoint drift problems appear in test and do not require plant outage time. Hence they did not appear in the COMPASS data.

When comparing the IC COMPASS data to other systems it is apparent that the reliability of the IC benefits from simplicity.