

TABLE 3.2-E

INSTRUMENTATION THAT MONITORS DRYWELL LEAK DETECTION

Minimum No. of Operable Instrument Channels	Instrument	No. of Instrument Channels Provided by Design	Action
1	Sump System (1)	6	(3)
1	Air Sampling System (2)	6	(3)

NOTES FOR TABLE 3.2-E

(1) The Sump System is comprised of the Equipment Drain Sump and Floor Drain Sump Sub-systems.

The Equipment Drain Sump Sub-system consists of one Equipment Drain Sump Flow Integrator and two Equipment Drain Sump Flow Timers. The Floor Drain Sump Sub-system likewise consists of one Floor Drain Sump Flow Integrator and two Floor Drain Sump Flow Timers. The Sump Sub-system is operable when any one of these six devices are operable.

(2) The Air Sampling System provides a backup system to the Sump System.

Action for Table 3.2-E

(3) See Specification 3.6.C.

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timer is set to annunciate before the values specified in Specification 3.6.C are exceeded. An air sampling system is also provided, as a backup to the sump system, to detect leakage inside the primary containment.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

Trip function settings are included for instrument a.c. and battery busses for surveillance of undervoltage relays. The undervoltage relays are required to sense a reduction in the power source voltage so that the subject instruments can be transferred to an alternate power source.

LIMITING CONDITION FOR OPERATIONSURVEILLANCE REQUIREMENTC. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:
 - a. 5 gpm unidentified leakage.
 - b. 2 gpm increase in unidentified leakage within a 24 hr. period.
 - c. 25 gpm total leakage.
2. The sump system shall be operable any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. From and after the date that the sump system is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 24 hours unless the system is made operable sooner, provided the air sampling system is operable.
3. If the conditions in 1 or 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, both safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.

C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump system and recorded at least once per day.
2. The air sampling system shall be checked and recorded at least once per day.

D. Safety and Relief Valves

1. At least one safety valve and 3 relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

The setpoint of the safety valves shall be as specified in Specification 2.2.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
<p>3. Following 1-pump operation, the discharge valve of the lower speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.</p>	
<p>6. <u>Structural Integrity</u></p> <p>The structural integrity of the pressure boundaries shall be maintained at the level required by the original acceptance standard throughout the life of the plant.</p>	<p>6. <u>Structural Integrity</u></p> <p>1. In-service inspection of ASME Code Class I, Class II and Class III components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).</p> <p>a. The second 10 year interval for the inservice inspection program described above commenced on November 1, 1985.</p> <p>2. In-service testing of ASME Code Class I, Class II and Class III pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).</p> <p>a. The second 10-year interval for the inservice testing program described above commenced on February 1, 1985.</p>

LIMITING CONDITION FOR OPERATIONSURVEILLANCE REQUIREMENT

3. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in this generic letter.

establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

Identified and unidentified leakage are defined in the DAEC Updated FSAR, Section 5.2.5.2.2. Total leakage is defined as the sum of identified and unidentified leakage.

The capacity of the drywell floor sump pumps is 50 gpm and the capacity of the drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with margin.

DAEC surveillance procedures require both identified and unidentified leakage to be determined at approximately 4 hour intervals. Should leakage exceed the allowed limits, control room alarms actuated by the equipment drain sump and floor drain sump pump timers are provided to indicate this condition, thus, continuous leakage detection capability is provided by design.

The requirement that an increase in unidentified leakage shall not exceed 2 gpm in a 24 hour period is based on maintaining the ability to detect small leaks in a reasonably short time such that corrective action can be initiated. However, during reactor startup and ascension to normal operating pressure, leakage should be closely monitored until normal operating pressure is achieved and a "baseline" leakage rate can be established to which any leakage increase can be compared.

The primary containment atmosphere radioactivity detector provides a sensitive and rapid indication of increased nuclear system leakage. The primary containment environment is continuously sampled from one of three locations which are chosen to provide both a representative gas mixture and an indication of the location of the leakage.

The sample air undergoes three separate processes in which the radioactive noble gas, halogen, and particulate contents are determined. This system is thus a three channel monitoring system. The processed air is returned to the drywell.

The primary containment atmosphere radioactivity detector serves as a sensitive, reliable backup to the other methods of leak detection. It is anticipated that the particulate detector will be the primary indication of leakage, with the halogen and noble gas detectors serving as indication of the primary containment environment if primary containment venting is required. These detectors in conjunction with an isotopic analysis can be used to indicate whether the detected leak is from a steam or water system. This system is not capable of accurately quantifying coolant leakage rates. Because the Air Sampling system is not capable of determining leak rate, it is considered a backup system to the sump system, and no LCO is associated with it. It is intended to be a compensatory measure used when the sump system is inoperable.

The first 10-year interval for inservice inspections in accordance with the ASME Boiler and Pressure Vessel Code, Section XI commenced on February 1, 1975. This interval was extended for 9 months because of a 9 month outage for replacement of recirculation system inlet nozzle safe-ends in 1978-79. Therefore, the first 10-year interval ended on October 31, 1985.

The second 10-year interval for inservice inspections commenced on November 1, 1985 and is scheduled to end on October 31, 1995. The second 10-year inspection program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications as stated in 10 CFR 50.55a.

Visual inspections for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC and is delineated by Section XI of the ASME Code. These studies show that it requires thousands of stress cycles at stresses beyond those expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results, only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is

proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Section 5.2.4 of the Updated FSAR provides details of the inservice inspection.

Starting with the Cycle 9/10 Refueling Outage, an augmented inspection program was implemented to address concerns relating to Intergranular Stress Corrosion Cracking (IGSCC) in reactor coolant piping made of austenitic stainless steel. The augmented inspection program conforms to the NRC staff's positions set forth in Generic Letter 88-01 and NUREG-0313, Revision 2 for inspection schedule, inspection methods and personnel, and inspection sample expansion.

The first 10-year interval for inservice testing of pumps and valves in accordance with the ASME Code, Section XI commenced on February 1, 1975 and ended on January 31, 1985. The second 10-year inservice testing interval commenced on February 1, 1985 and is scheduled to end on January 31, 1995. The second 10-year testing program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications of 10 CFR 50.55a. Section 3.9.6 of the Updated FSAR describes the inservice testing program.

3.6.H & 4.6.H BASES:

Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or other severe transient, while accommodating normal thermal motion during system startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of damage to piping as a result of a seismic or other event initiating dynamic loads or, in the case of a frozen snubber, exceeding allowable stress limits during system thermal transients. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Safety Analysis

1.0 Introduction

By letter dated July 27, 1988 from W. Rothert to T. Murley, Iowa Electric Light and Power Company requested changes to the Duane Arnold Energy Center (DAEC) Technical Specifications (TSs) to incorporate requirements for an augmented ISI program and revise provisions concerning the leakage detection system to meet the requirements of NRC Generic Letter 88-01 and NUREG-0313, Revision 2.

In addition, changes to the DAEC TSs to reflect the beginning dates for the second ten-year intervals for the Inservice Inspection (ISI) and Inservice Testing (IST) programs. An interim requirement for inspections of the recirculation system inlet nozzle safe-ends has been deleted and these will now be covered by the augmented ISI program.

2.0 Evaluation

The augmented ISI program for the DAEC was separately transmitted by letter dated July 27, 1988. To ensure that no defects develop in the Reactor Coolant Pressure Boundary (RCPB) during service, selected welds and weld heat-affected zones will be inspected periodically throughout the remaining life of the plant. The augmented ISI program will ensure these periodic inspections are conducted in accordance with the requirements of Generic Letter 88-01 and NUREG-0313, Revision 2. In addition, the reduction in the allowed outage time for inoperable equipment associated with the coolant leakage detection system provides added conservatism in the Technical Specifications and is in accordance with the staffs position in NRC Generic Letter 88-01.

These changes provide reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of the RCPB is compromised.

Therefore, we conclude that these changes correspond to the staff positions in NRC Generic Letter 88-01 and NUREG-0313, Revision 2, and are acceptable.

The remaining proposed changes to the DAEC TSs are considered to be editorial in nature or to explain certain surveillance requirements associated with the ISI and IST programs at the DAEC. We conclude that these proposed changes are also acceptable.