REMOTE SHUTDOWN INSTRUMENTATION (Continued)

HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the control room WILL BE ~~emergency ventilation system will~~ automatically isolated ~~the control room and initiate its operation~~ in the recirculation mode to provide the required protection. The chlorine detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operations Against an Accidental Chlorine Release," February 1975.

HE CONTROL
OM VENTILATION
STEM WILL BE
NUALLY
STARTED

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

Operability of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Docket No. 50-346

License No. NPF-3

Serial No. 738

September 25, 1981

Attachment 3

I. Change to Davis-Besse Nuclear Power Station Unit 1, Appendix A, Technical Specifications 4.4.6.2.2 and 6.9.1.9.

A. Time Required to Implement

This change is to be effective upon NRC approval.

B. Reason for Change (Facility Change Request 81-143)

To combine leakage test without opening the motor operated containment isolation valves and delete the reporting requirements when any part of the system is inoperable only for surveillance testing, instrument calibration or preventive maintenance purposes.

C. Safety Evaluation

See attached.

bj a/17

Safety Evaluation

This Amendment Request proposes changes to Sections 4.4.6.2.3 and 6.9.1.9 of the Davis-Besse Technical Specifications. These Technical Specifications relate to the reactor coolant system pressure isolation valves CF30, DH76 (CF31, DH77) and the thirty day reporting requirements respectively.

The safety function of the pressure isolation valves CF30, DH76 (CF31, DH77) and containment isolation valve DH1A (DH1B) is to provide a pressure isolation barrier between the high pressure reactor coolant system inside the containment and the low pressure Decay Heat Removal System outside the containment. Whenever the integrity of a pressure isolation valve listed in Table 3.4-2 cannot be demonstrated, Toledo Edison will determine the integrity of the high pressure line by performing either a leakage test of the remaining pressure isolation valve or a combined leakage test of the remaining pressure isolation valve in series with the closed motor operated containment isolation valve. The combined leakage test can be done when the reactor is in mode 3 during shut down or startup. If at least one of the two pressure isolation valves, DH76 or CF30 (DH77 or CF31) is not leaking and the motor operated containment isolation valve, DH1A (DH1B) is closed before the combined leakage test is performed, the integrity of the low pressure line is not challenged. Moreover, the plant is only allowed to stay in this configuration (i.e. with DH1A or DH1B closed) for 72 hours per Technical Specifications Section 3.5.2. Therefore, it is concluded that the combined leakage test will not compromise the safety of the plant.

The safety function of the affected reporting requirements is to make the Nuclear Regulatory Commission (NRC) aware of any situations which require entry into the action statements per Technical Specifications. During routine surveillance testing, instrument calibration, or preventive maintenance the plant configurations are sometimes rendered less conservative than those established by the Technical Specifications, or the station is made to operate in a degraded mode permitted by a limiting condition for operation. When one of these two conditions exist, the plant will be placed in the action statements. Consequently, a Licensee Event Report (LER) will have to be written and submitted to the NRC per Technical Specification Section 6.9.1.9.

Toledo Edison feels that this reporting requirement is not applicable for the purpose of surveillance testing, instrument calibration, or preventive maintenance. Regulatory Guide 1.16 also supports this exception to the reporting requirement. This Amendment request proposes to incorporate this Regulatory Guide 1.16 exemption clause into the Technical Specifications. This change proposes to delete the reporting requirement when the plant is placed in the action statements due to routine surveillance testing, instrument calibration, or preventive maintenance. Compliance with other existing reporting requirements is not affected by this change.

Pursuant to the above, it is concluded that the proposed changes do not involve an unreviewed safety question.

bj a/18

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals to the makeup system when the Reactor Coolant System pressure is 2185 ± 20 psig at least once per 31 days.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-2 shall be individually demonstrated OPERABLE by verifying leakage testing (or the equivalent) to be within its limit prior to entering MODE 2:

a. After each refueling outage,

b. Whenever the plant has been in COLD SHUTDOWN for 72 hours, or more, and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

4.4.6.2.3 Whenever integrity of a pressure isolation valve listed in Table 3.4-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the ~~other~~ closed valve located in the high pressure piping shall be recorded daily.

OR THE INTEGRITY OF THE REMAINING PRESSU[RE] ISOLATION VALVE IN SERIES WITH THE MOTOR-OPERATED CONTAINMENT ISOLATION VALV[E]

MOTOR OPERATED CONTAINMENT ISOLATION

WHERE TEST RESULTS THEMSELVES REVEAL A DEGRADED CONDITION REQUIRING CORRECTIVE ACTION

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS*

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

* ROUTINE SURVEILLANCE TESTING, INSTRUMENT CALIBRATION, OR PREVENTIVE MAINTENANCE WHICH REQUIRE SYSTEM CONFIGURATIONS AS DESCRIBED IN SECTION 6.9.1.9a and 6.9.1.9b NEED NOT BE REPORTED EXCEPT

Docket No. 50-346

License No. NPF-3

Serial No. 738

September 25, 1981

Attachment 4

I. Change to Davis-Besse Nuclear Power Station Unit 1, Appendix A, Technical Specifications 3.4.8, Table 4.4-4, Figure 3.4-1 and Bases.

A. Time Required to Implement

This change is to be effective upon NRC approval.

B. Reason for Change (Facility Change Request 81-163, Rev. A)

The Technical Specifications values for iodine are currently based upon a parametric evaluation by the NRC of typical site location. The values are conservative in the specific site parameters of the Davis-Besse site. Operating experience shows these values to be conservative and when the plant trips, the iodine spike exceeds the limit for Dose Equivalent I-131. Changing the Dose Equivalent limit for I-131 will not exceed 10CFR100 dose limits and will reduce Licensee Event Reports on a normal reactor trip event.

C. Safety Evaluation

See attached.

bj a/20

Safety Evaluation

This amendment request proposes a change to the dose equivalent I-131 concentration in the reactor coolant system from 1.0 uci/gm to 3.2 uci/gm. The Technical Specifications limitation on the Dose Equivalent I-131 concentration in the primary coolant is established to insure that the resulting two hour doses at the site boundary will not exceed a small fraction of the 10CFR100 limit following a design basis steam generator tube rupture accident in conjunction with an assumed steady state primary to secondary leakage rate of 1.0 gpm. The two hour dose at the site boundary for 3.2 uci/gm dose equivalent I-131 has already been analyzed in FSAR Section 15.4.2.3 for the design basis steam generator tube rupture accident. The resultant 0.23 Rem whole body and 27.1 Rem thyroid dose as presented in FSAR Table 15.4.2-3 are below one tenth of the 10CFR100 limits of 25 Rem for whole body and 300 Rem for thyroid. The FSAR doses are calculated assuming 1% defective fuel rods operated at steady state with 1 gpm primary to secondary leakage rate prior to the postulated double-ended rupture accident of one steam generator tube. The resulting site boundary doses from a less severe rupture should be much lower than the case mentioned above since the activity released is far less in magnitude. Based on the above, it is concluded that the original premise of doses being a small fraction of 10CFR100 limits is still met.

Following the implementation of the proposed change, the primary coolant specific activity sampling analysis program as defined in Technical Specifications Table 4.4-4 will still be complied with as required.

Therefore, it is concluded that no unreviewed safety question is involved.

bj a/21

Markups of changes in T.S.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ ~~1.0~~ 3.2 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1, 2 and 3*.

a. With the specific activity of the primary coolant > ~~1.0~~ 3.2 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10% of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > ~~1.0~~ 3.2 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} < 530°F within 6 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in at least HOT STANDBY with T_{avg} < 530°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant > ~~1.0~~ 3.2 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

*With $T_{avg} \geq$ 530°F.

ACTION: (Continued)

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded ~~1.0~~ 3.2 μCi/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once each 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\overline{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 3.2 ~~1.0~~ μCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

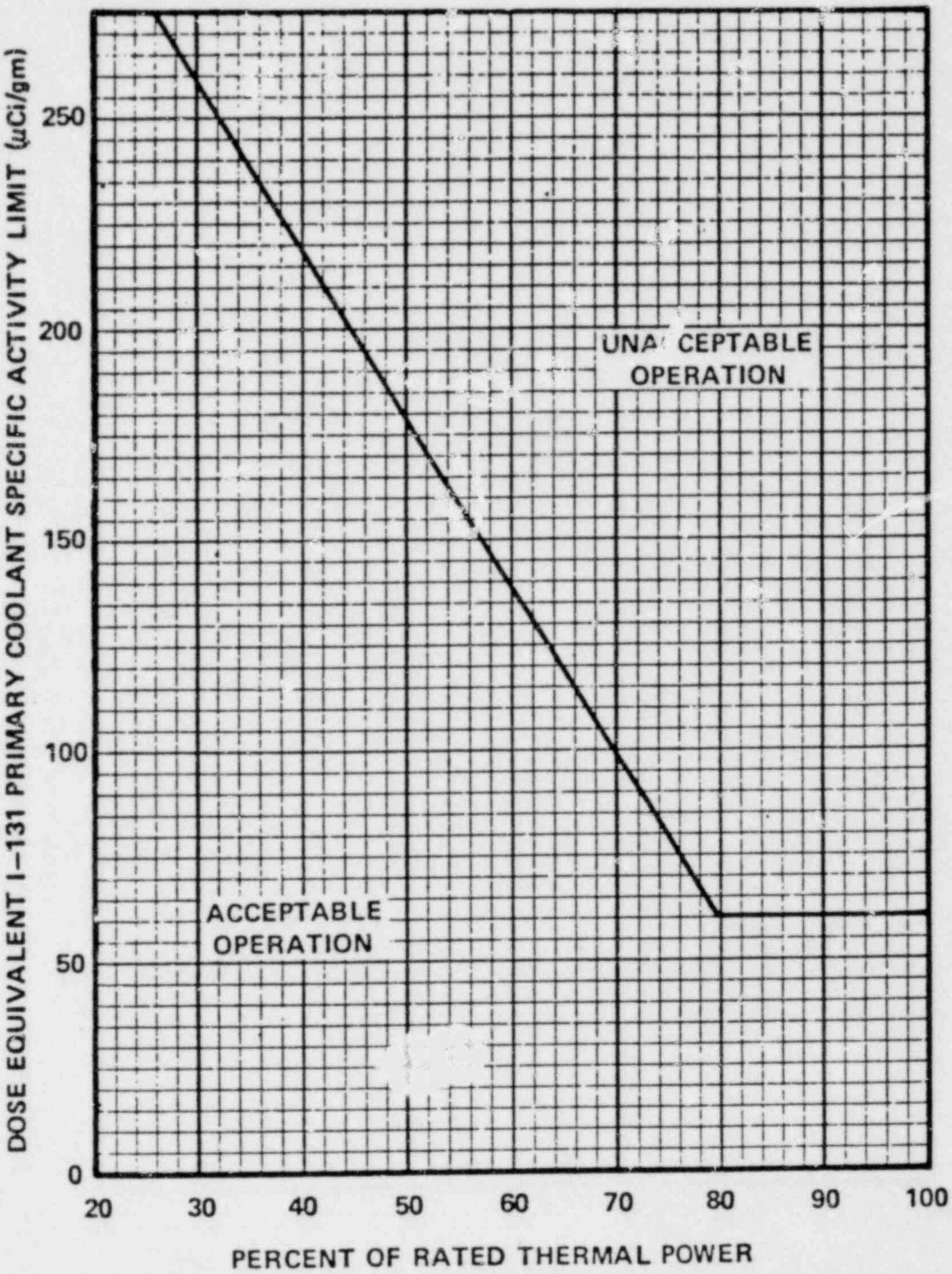

FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > ~~1.0~~ 3.2 μCi/gram Dose Equivalent I-131

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits shown on Table 3.4-1 provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of the Part 100 limit following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in the specific site parameters of the site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

BASES

3.2

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > ~~1.0~~ 3.2 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding ~~1.0~~ μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the units yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to $\leq$ 530°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The pressure-temperature limits of the reactor coolant pressure boundary are established in accordance with the requirements of Appendix G to 10 CFR 50 and with the thermal and loading cycles used for design purposes.

The limitations prevent non-ductile failure during normal operation, including anticipated operational occurrences and system hydrostatic tests. The limits also prevent exceeding stress limits during cyclic operation. The loading conditions of interest include:

1. Normal operations, including heatup and cooldown,
2. Inservice leak and hydrostatic tests, and
3. Reactor core operation.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. The closure head region, reactor vessel outlet nozzles and the beltline region have been identified to be the only regions of the reactor vessel, and consequently of the reactor coolant pressure boundary, that determine the pressure-temperature limitations concerning non-ductile failure.