TOLEDO EDISON

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Docket No. 50-346

License No. NPF-3

Serial No. 1266

September 26, 1986

Mr. John F. Stolz, Director
PWR Project Directorate No. 6
Division of PWR Licensing-B
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

This is in response to your letter dated October 16, 1985 (Log No. 1838) concerning conformance to Regulatory Guide 1.97 by the Davis-Besse Nuclear Power Station, Unit No. 1. Your letter transmitted an interim report on the NRC's evaluation of Davis-Besse's instrumentation as compared to the recommendation of Regulatory Guide 1.97, Revision 3 and requested a response to the open items identified in the report. Toledo Edison provided a response dated January 27, 1986 (Serial No. 1232) for all items except 3.3.30/4.17. Attached is the response to item 3.3.30/4.17 which completes our response concerning conformance to Regulatory Guide 1.97.

Very truly yours,

Joe Williams, Jr. /ABA

JW:MLB:GAB:plf

Attachment

cc: DB-1 NRC Resident Inspector

A007
1/1

NRC Position

3.3.30/4.17

Noble Gas Vent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves

Regulatory Guide 1.97 recommends this Category 2 instrumentation for the variable. The recommended parameters to be monitored for this variable are noble gas, duration of release in seconds and mass of steam per unit item.

In regard to this variable, the licensee states that the position indication of the atmospheric vent valves (AVV) and the main steam safety valves (MSSV) is not monitored and that monitoring is not required to mitigate the consequences of a design basis accident. AVV position is indicated in the control room via indicating lights. In addition, the licensee states that the sound emitted from the valves provides an audible indication to the operators when either the MSSVs or AVVs lift. Dose estimate procedures are used to quantify noble gas/radioiodine releases from the AVVs, MSSVs and auxiliary steam turbine exhaust utilizing the currently installed main steamline radiation monitors or the steam jet air ejector radiation monitor.

We find this arrangement unacceptable for this variable. First, the licensee should identify the range of the main steamline monitors, verify that the range is adequate and that the instrumentation is Category 2. Second, the licensee should indicate how the duration of the release and the mass of steam per unit time is determined. Third, the licensee should show that the results derived from this method are within an acceptable tolerance from the actual release.

Response

The indicated range for the existing main steam line radiation monitors is 10 to 10^6 CPM. These monitors have two modes of operation, the "analyze" and "gross" mode. In the analyze mode the monitors will detect specific isotopes. The main steam line radiation monitors are normally operated in the analyze mode and calibrated to detect N-16 which is the first reactor coolant radioisotope that will be detectable in the main steam line if a Steam Generator Tube Rupture (SGTR) or leak occurs. With the monitors set in this configuration, the 10 to 10^6 CPM range corresponds to an activity of approximately 1.1×10^{-7} to 1.1×10^{-2} μCi/cc in the steam line (see USAR Table 11.4-1).

For release assessment purposes the monitors are placed in the gross mode. In the gross mode of operation the monitors will detect any isotopes with energy levels above 80 Kev and up to approximately 2 Mev. With the monitors set in this configuration the 10 to 10^6 CPM range corresponds to an activity of approximately 9×10^{-5} to 9 μCi/cc in the steam line.

The sensitivity of the radiation monitor (CPM per μCi/cc) is calculated using the following method. An attenuated calibration curve for these monitors was developed based on a steam line wall thickness of 2.375" and utilizing the shielding buildup factor for the pipe metal. The energy spectrum for radioactivity present in the main steam line at different times after shutdown was calculated using noble gas fission product activities applicable to normal power operation at 0.1% failed fuel (USAR Table 11.1-4). The sensitivity of the radiation monitors (CPM per μCi/cc) with respect to time was calculated using the energy spectra, the attenuated calibration curve and the detected source volume. The response of the radiation monitors is mainly due to the high energy gamma radiation from the short-lived noble gas isotopes. The most abundant Xe-133 isotope (5.2 day half-life) would not contribute to the detector response because the low energy gamma radiation for Xe-133 would be attenuated in the steam line pipe wall. Due to the decay of short-lived isotopes the detector sensitivity (specific activity in the steam line corresponding to a count rate at the detector) changes with time during several hours following the event. For example, the activity range the radiation monitors can detect at the beginning of an accident is 9×10^{-5} to 9 μCi/cc, whereas at 8 hours after the beginning of the accident the detectable activity range will be 10^{-3} to 10^{2} μCi/cc.

The design basis event during which this monitor is required to function is a steam generator tube rupture accident. For a double ended tube rupture accident the initial reactor coolant leak rate is approximately 400 gpm (39 lb/sec). This leak would be diluted in a steam flow rate of 1600 lb/sec in one steam generator. Thus, for a reactor coolant noble gas concentration of 26 μCi/gm, the concentration in the steam line during initial phases of the accident would be 2.4×10^{-2} μCi/cc. As the accident progresses, the auxiliary feedwater and main feedwater feeding the steam generator with the tube rupture would be isolated. Normally, the reactor will be cooled by steaming the good steam generator. However, under certain conditions (primary to secondary leak <50 gpm) the reactor may also be cooled by steaming the bad steam generator. Assuming that the steaming rate from the affected steam generator is the same as the reactor coolant leak rate, an upper bound for the steam concentration can be calculated. For a reactor coolant noble gas concentration of 26 μCi/gm (corresponding to 0.1% failed fuel) the upper bound steam noble gas concentration would be 1 μCi/cc. A 0.1% failed fuel condition approximately corresponds to a RCS concentration 0.45 μCi/gm dose equivalent of Iodine - 131. Thus, if the reactor is operating at a technical specification limit of 1 μCi/gm dose equivalent of Iodine - 131, the corresponding reactor coolant noble gas concentration would be 58 μCi/gm. The upper bound steam line noble gas concentration for this condition would be 2.2 μCi/cc. The existing monitor range envelopes the upper bound steam line concentrations even when the reactor is operating at the technical specification limit for failed fuel. Since the sensitivity of the monitors is based on only the noble gases present in the reactor coolant, the presence of any additional isotopes in the steam (e.g., iodine partitioned in the steam generator) would tend to overestimate the releases.

Conversations with the radiation monitors vendor indicated that the accuracy for the monitor with the type of configuration used at Davis-Besse should be less than ±50%. Thus, it is expected that the overall system accuracy for the monitors would be within a factor of 2 as recommended in the Regulatory Guide 1.97.

The main steamline radiation monitors are utilized to detect and monitor a SGTR or leak. They are not required to mitigate the consequences of a high energy line break (HELB) or loss of coolant accident (LOCA) and, as such, need not be qualified to do so. The harsh environment created by a SGTR or leak is radiation. The environmental qualification of the main steam line radiation monitors have been addressed per 10 CFR 50.49, with respect to radiation, and found acceptable. Continuous indication is available to the operators via computer points and trend recorders. Analog indication is available in the control room cabinet room. Therefore, the radiation monitors meet the Category 2 requirements.

The radiation monitor readings, in conjunction with the steam flow rates, are used in the determination of emergency protective actions. The existing procedures assume that the entire steam flow leaving the MSSVs and AVVs is at the concentrations seen by the radiation monitors. The steam flow rate corresponds to the feedwater flow required to remove the decay heat. In reality, the good steam generator will be used to cool the plant as quickly as possible so that the affected steam generator can be isolated. The use of these conservative assumptions coupled with the reactor coolant activity within the technical specification limits in the Emergency Planning calculation may result in recommending the lowest level of protective action (sheltering) when it may not be required. This emergency action is considered acceptable in light of the conservative assumptions used. Based on the above discussion, it is considered the results derived using radiation monitor readings are within acceptable tolerance from the actual release. Thus, we meet the intent of Regulatory Guide 1.97 using the existing instrumentation and without providing the radiation monitor range recommended in NUREG 0737 for operating plants.

To more accurately reflect the steam flow rates and, hence, release rates as noted above, Toledo Edison has currently scheduled (see letter dated January 27, 1986, Serial No. 1232) to install the MSSV and AVV position indications during the fifth (next) refueling outage. The duration of the release and the mass of steam per unit time will be determined by a computer program which will be continuously monitoring several plant parameters. The following plant parameters are presently being considered for inputs into the computer program:

MSSV and AVV	Position indication - currently scheduled to be installed in the fifth (next) refueling outage in accordance with our Integrated Implementation Plan

SG 1 and 2	Outlet Pressure
Main Steam Line 1 and 2	Radiation Monitors
AFPT 1 and 2	Isolation Valves position indication
AFPT 1 and 2	Steam Admission Valve position indication
AFPT 1 and 2	Stop Valve position indication
AFPT 1 and 2	Speed indication

The MSSV and AVV position indication combined with the SG outlet pressure will quantify the mass of steam released from the main steam line connected to the SG with the tube rupture or leak. The affected SG can be identified via the difference in the indications of the main steam line radiation monitors. The AFPT isolation valve, steam admission valve and stop valve position indication will identify which AFPT is lined up to the effected SG. A curve will be developed to quantify the mass steam flow vs. AFPT speed from the turbine manufacturers data.

This new method of determining the mass of steam released will more closely quantify the mass of steam released per unit time with or without the MSIV's being closed, and will also reduce the extreme conservatism presently found in our procedure for dose rate (steam flow x steam radiation concentration) which was originally based upon Steam Generator steam flow which assumed all of the steam is released to the atmosphere. This method of determining the duration of the release and the mass of steam per unit time will be incorporated into EP 1202.01, Steam Generator Tube Rupture, AB 1203.40, Steam Generator Tube Leak and AD 1827.10, Emergency Off-Site Dose Estimate (these procedures will be revised after the required modifications are completed in accordance with our Integrated Implementation Plan as discussed in our letter dated January 27, 1986, Serial No. 1232). In addition to the conservative calculation performed to assess the activity and the extent of the release via AD 1827.10, EP 1202.01 and AB 1203.40, AD 1805.05, Radiation Monitoring Team Surveys, is performed on-site and off-site to obtain actual radiation levels due to the release.

Therefore, based on the above, Toledo Edison meets the intent of Regulatory Guide 1.97, Revision 3 for "Noble Gas from Steam Generator Safety Relief Valves or Atmospheric Dump Valves".

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TABLE 11.4-1

Liquid, Gas, and Airborne Radiation Monitors

LIQUID

System Designation	Required Measurements	Sensitivity μCi/cc	Range μCi/cc (CPM)	Background (mr/hr)	Type Detector	Flowrates (gpm) Monitored Process Stream / Respective Unit Effluent
Reactor coolant (Failed fuel Detector) RE-1998	Cs-138 and gross gamma	2.3×10^{-6}	1.64×10^{-7} to 1.64×10^{-2}* (10 to 10^{6})	100	Gamma scintillation off-line (snow plow)	150/none
Misc. radwaste system outlet RE-1878A	gross radio-activity	2.02×10^{-6}	1.64×10^{-7} to 1.64×10^{-2} (10 to 10^{6})	100	Gamma scintillation off-line	70/20000
RE-1878B	gross radio-activity	1.54×10^{-6}	1.2×10^{-7} to 1.2×10^{-2} (10 to 10^{6})	100	Gamma scintillation off-line	70/20000
Clean radwaste system outlet RE-1770A	gross radio-activity	4×10^{-7}	1.38×10^{-7} to 1.38×10^{-2} (10 to 10^{6})	10	Gamma scintillation off-line	70/20000
RE-1770B	gross radio-activity	3×10^{-7}	1.04×10^{-7} to 1.04×10^{-2} (10 to 10^{6})	10	Gamma scintillation off-line	70/20000
Component Cooling Return line RE-1412 RE-1413	gross radio-activity	1.01×10^{-6}	2.3×10^{-7} to 2.3×10^{-2} (10 to 10^{6})	0.5	Gamma scintillation off-line (snow plow)	7900/none
Main steam line RE-600 RE-609	N-16	0.5×10^{-6}	1.1×10^{-7} to 1.1×10^{-2} (10 to 10^{6})	0.5	Gamma scintillation off-line (snow plow)	5.9×10^{6} lbs./hr/none
Service water outlet RE-8432	gross radio-activity	0.54×10^{-6}	1.2×10^{-7} to 1.2×10^{-2} (10 to 10^{6})	Negligible	Gamma scintillation off-line	20000

* Provided with a collimator to change the range to 1.64×10^{-3} – $1.64 \times 10^{+2}$ μCi/cc. This collimator may be installed if a large amount of failed fuel occurs.

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TABLE 11.1-4

Calculated Reactor Coolant Activity For 0.1% Failed Fuel

Radionuclide	Fission Product Activity in Reactor Coolant, μCi/cc*, For Indicated Time in Equilibrium Cycle, Full Power Days											
	4	10	30	60	90	120	150	180	210	240	267	277
Kr-83m	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.05(-2)	2.04(-2)	2.05(-2)
Kr-85m	0.109	0.109	0.109	0.109	0.109	0.108	0.108	0.108	0.108	0.108	0.108	0.109
Kr-85	6.74(-3)	1.68(-2)	5.03(-2)	9.92(-2)	0.144	0.182	0.211	0.224	0.221	0.184	0.106	0.132
Kr-87	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	5.93(-2)	5.96(-2)
Kr-88	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.190	0.191
Rb-88	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192
Sr-89	2.27(-4)	2.38(-4)	2.58(-4)	2.80(-4)	2.94(-4)	3.04(-4)	3.10(-4)	3.15(-4)	3.18(-4)	3.20(-4)	3.21(-4)	3.21(-4)
Sr-90	5.19(-6)	5.40(-6)	5.78(-6)	6.34(-6)	6.90(-6)	7.46(-6)	8.02(-6)	8.58(-6)	9.13(-6)	9.68(-6)	1.02(-5)	1.04(-5)
Sr-91	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)	2.03(-3)
Sr-92	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)	6.21(-4)
Y-90	9.78(-6)	3.58(-5)	1.36(-4)	2.86(-4)	4.26(-4)	5.52(-4)	6.54(-4)	7.11(-4)	7.19(-4)	6.16(-4)	3.59(-4)	1.72(-4)
Y-91	2.24(-4)	5.76(-4)	1.58(-3)	2.69(-3)	3.40(-3)	3.83(-3)	4.03(-3)	3.97(-3)	3.73(-3)	3.12(-3)	1.98(-3)	1.07(-3)
Mo-99	7.86(-2)	0.209	0.292	0.293	0.292	0.292	0.290	0.288	0.285	0.277	0.257	0.209
Xe-131m	1.60(-2)	3.41(-2)	8.70(-2)	0.140	0.157	0.160	0.158	0.153	0.147	0.133	0.104	0.134
Xe-133m	4.67(-2)	0.150	0.200	0.200	0.200	0.199	0.198	0.197	0.195	0.191	0.179	0.202
Xe-133	1.84	6.82	16.2	17.7	17.6	17.5	17.4	17.1	16.8	16.0	14.0	17.1
Xe-135m	6.58(-2)	6.59(-2)	6.59(-2)	6.59(-2)	6.59(-2)	6.59(-2)	6.58(-2)	6.58(-2)	6.57(-2)	6.56(-2)	6.51(-2)	6.60(-2)
Xe-135	0.360	0.361	0.361	0.361	0.361	0.361	0.360	0.360	0.359	0.357	0.352	0.362
Xe-138	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)	3.64(-2)
I-131	8.58(-2)	0.145	0.215	0.229	0.230	0.230	0.230	0.230	0.230	0.230	0.230	0.230
I-132	5.26(-2)	0.114	0.159	0.160	0.160	0.160	0.160	0.160	0.160	0.160	0.160	0.160
I-133	0.250	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269
I-134	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)	3.26(-2)
I-135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135
Cs-134	8.42(-3)	2.09(-2)	6.18(-2)	0.121	0.174	0.223	0.263	0.286	0.291	0.255	0.157	8.10(-2)
Cs-136	4.38(-3)	1.17(-2)	3.29(-2)	4.82(-2)	5.22(-2)	5.27(-2)	5.21(-2)	5.05(-2)	4.85(-2)	4.41(-2)	3.46(-2)	2.26(-2)
Cs-137	2.59(-2)	6.45(-2)	0.193	0.380	0.554	0.710	0.834	0.900	0.904	0.773	0.459	0.230
Cs-138	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.23(-2)	5.22(-2)	5.21(-2)
Ba-137m	2.37(-2)	5.93(-2)	0.177	0.350	0.510	0.653	0.767	0.828	0.832	0.711	0.422	0.212
Ba-139	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)	5.41(-3)
Ba-140	1.92(-4)	2.52(-4)	3.51(-4)	3.91(-4)	3.99(-4)	4.01(-4)	4.01(-4)	4.01(-4)	4.01(-4)	4.01(-4)	4.01(-4)	4.01(-4)
La-140	6.86(-5)	9.44(-5)	1.38(-4)	1.56(-4)	1.59(-4)	1.60(-4)	1.60(-4)	1.60(-4)	1.60(-4)	1.60(-4)	1.60(-4)	1.60(-4)
Ce-144	2.08(-5)	2.16(-5)	2.31(-5)	2.53(-5)	2.73(-5)	2.91(-5)	3.08(-5)	3.25(-5)	3.40(-5)	3.54(-5)	3.65(-5)	3.69(-5)

*Notes: 1 - Coolant Density (ρ) = 0.713 gm/cc

2 - In this table, a number $A \times 10^{-B}$ is expressed as A(-B)