Attachment 3

BWR OWNERS' GROUP

T. J. Dente, Chairman

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BWROG-8222

U. S. Nuclear Regulatory Commission
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Darrell G. Eisenhut, Director

Gentlemen:

SUBJECT: SUPPLEMENT TO BWR OWNERS GROUP EVALUATION OF NUREG-0737, ITEM II.E.4.2(7)

References:

1) Letter from T. J. Dente (BWR Owners Group) to D. G. Eisenhut (NRC) titled "BWR Owners Group Evaluation of NUREG-0737, Item II.E.4.2(7)," dated June 29, 1981
2) Letter from D. G. Eisenhut (NRC) to T. J. Dente (BWROG) titled "NUREG-0737, Item II.E.4.2(7): Containment Isolation Dependability - Isolation on High Radiation," dated October 14, 1981
3) Minutes of Meeting between NRC Staff and GE/BWROG on November 19, 1981
4) Telecon between F. Hayes (GE) and D. Verrelli (NRC) on February 18, 1982

Reference 1 transmitted the results of the BWR Owners Group initial evaluation of NUREG-0737, Item II.E.4.2, Part 7. That evaluation concluded that automatic isolation of the containment vent and purge valves on high containment radiation is not necessary for Mark I and II plants. The basis for that conclusion is that: (1) the vent and purge valves are normally closed, (2) there already exist separate and diverse signals for automatic closure of those valves, (3) there are various signals which will alert the operator to manually close the valves, and (4) the radiological consequence from a break sufficiently small so as not to automatically isolate these valves is acceptably low.

The NRC rejected the Owners Group position in Reference 2.

In response to the NRC rejection, the BWR Owners Group met with the NRC (Reference 3) in order to obtain a clarification of the principal NRC concerns that prompted this requirement as well as a clarification of the requirement itself. A presentation was given to the Staff by the BWR Owners Group in which it was shown that the benefit of adding an automatic radiation signal is negligible due to the existing BWR design capability. The principal concerns expressed by the NRC in Reference 2 were also addressed.

The NRC bases for implementation of II.E.4.2(7) as expressed at the meeting were: (1) the radiation signal would provide additional safety margin, (2) additional redundancy would be provided if one of the existing signals should fail, and (3) there would be additional protection against low rates of reactor coolant leakage which would not initiate an automatic signal with the current design.

The first two concerns were addressed by the Owners Group presentation on November 19, 1981. That presentation showed that the reliability of the existing isolation signals is extremely high. It also showed that the probability that automatic isolation on high radiation would be required is extremely low (10^{-5} for low dose consequence events and 10^{-9} to 10^{-12} for high dose consequence events). Such a small benefit would need to be weighed against the high cost of implementation (at least $500,000 per plant for a safety grade system).

With regard to the third NRC concern, the BWR Owners Group has conducted an evaluation for a typical plant of the radiological consequences of the limiting reactor coolant system break which would not result in automatic containment isolation for the current design. The performance of this analysis was discussed with the NRC per Reference 4. The result is an offsite thyroid dose on the order of .01 Rem which is well below the EPA's Protective Action Guide.

A list of the key input assumptions for that analysis is provided in Attachment 1. A discussion of the analytical procedure is provided in Attachment 2. These assumptions and methods will be provided to individual utilities for their use in performing plant unique analyses of the dose consequences for the limiting break. The NRC Staff has recommended the use of the EPA's Protective Action Guide as the acceptance criterion for this event (Reference 3). The BWR Owners Group believes that the dose limit of the EPA's Protective Action Guide is excessively restrictive to be used as a decision basis for installation of an additional automatic isolation signal for low probability events. However, after reviewing the margin against this criterion for our typical plant analysis, the Owners Group is willing to apply it in this instance to demonstrate that there is no need for automating the high radiation isolation signal.

If a plant unique analysis should show that offsite doses are in excess of this criterion, that utility may elect to adopt a more restrictive technical specification limit on primary coolant iodine concentration for that plant during venting and purging operations so that the acceptance criterion is satisfied. This would be a suitable alternate approach to satisfying the intent of NUREG-0737, Item II.E.4.2, Part 7, in lieu of an installation of an automatic high radiation isolation signal.

In conclusion, the calculation of acceptable offsite doses for the limiting break as defined herein, using the input assumptions and analysis methods defined in Attachments 1 and 2, is an acceptable alternative to installation of an automatic high radiation isolation signal. Results of a calculation for a typical plant show that this alternate approach is feasible.

Enclosed are copies of relevant correspondence on this subject.

The submittal of an Owners' Group position developed in response to an NRC requirement does not indicate that the Owners' Group unanimously endorses that position; rather, it indicates that a substantial number of members believe the position is responsive to the NRC requirement and adequately satisfies the requirement. Each member must formally endorse a position so developed and submitted in order for the position to become the member's position.

Very truly yours,

Thomas J. Dente

T. J. Dente, Chairman
BWR Owners' Group

TJD:rm/A04303

Enclosure

cc: BWR Owners' Group
J. F. Schilder (GE)
S. J. Stark (GE)
D. M. Verrelli (NRC)
W. R. Butler (NRC)
W. Pasedag (NRC)
V. Stello (NRC)

ATTACHMENT 1

Key Analysis Assumptions

1. Drywell pressure equal to containment isolation setpoint.
2. Drywell atmosphere is saturated steam.
3. No plateout or fallout of iodine in containment or vent piping.
4. No steam condensation in purge or vent pipes.
5. Break fluid is saturated water at 1000 psia (constant throughout the event).
6. All iodine in flashed coolant assumed released.
7. No credit for standby gas treatment system (SGTS) or reactor water cleanup system (RWCS) filtration.
8. Initial primary coolant iodine concentration at tech spec limit.
9. Iodine spiking included (95% cumulative probability value) for depressurization event.
10. Operator action time to close purge and vent valves = 10 minutes.
11. Annual average meteorology.
12. Regulatory Guide 1.3 breathing rates.

* Conservative assumption, since maximum leaks are on the order of 500 gpm and would be detectable in minutes.

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4/30/82

ATTACHMENT 2

Steam Leakage Rate Calculation

The steam leakage rate through the purge or vent line where the pressure drop is small relative to the inlet pressure is:

$$q = \left\{ \frac{\left[992.1\ \bar{v}_g\ d_1^4 \right] \left[(p_1 - p_2) - \left(\frac{z_1 - z_2}{144\ \bar{v}_g} \right) \right]}{\left[K_1 + \frac{v_{g2}}{\bar{v}_g} \left(\frac{d_1}{d_2} \right)^4 - \frac{v_{g1}}{\bar{v}_g} \right]} \right\}^{1/2}$$

where:

q = volumetric flow rate at $\bar{v}_g$, cfm.

d = internal diameter of line, inch.

p = pressure, psia.

v_g = specific volume of steam, ft^3/lb_m.

z = elevation above reference plane, ft.

K_1 = resistance coefficient of the line with respect to diameter, d_1.

$\bar{v}_g$ = $(v_{g1} + v_{g2})/2$

and the subscripts "1" and "2" refer to the conditions at the inlet and outlet of the line respectively.

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Break Flow Rate Calculation

The flow rate of high pressure saturated water required to produce the quantity of steam, q is

$$Q_b = \frac{7.4805\, v_{f3}}{\bar{v}_g\, x}\, q \qquad (2)$$

where

$$x = \frac{h_{f3} - h_{f1}}{h_{fg}} \qquad (3)$$

and

Q_b = break flow rate at v_{f3}, gpm.

x = quality, flashing fraction.

v_f = specific volume of liquid, ft^3/lb_m.

h_f = specific enthalpy of liquid, Btu/lb_m.

h_{fg} = specific enthalpy of vaporization, Btu/lb_m.

and subscript "3" refers to conditions in the primary coolant line.

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Calculation of Activity of Leaking Steam

The activity of the steam leaving the vent or purge line at a given time which was produced by the flashing of the high pressure saturated water with activity, a is

$$\dot{A} = \frac{a f_p (1 - \epsilon_f)}{36.74\, \bar{v}_g} q \qquad (4)$$

where:

$\dot{A}$ = activity leaving the line, Ci/hr.

a = activity of high pressure saturated water, μCi/gram.

f_p = partitioning factor = $\frac{\text{activity of flashed steam per gram}}{\text{activity of high pressure saturated liquid per gram}}$

ε_f = filter efficiency.

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Dose Calculations

The offsite dose calculation is as follows:

$$\Delta D_i = \Delta A_i \left(\frac{X}{Q}\right)_i B_i K$$

where:

ΔD_i = Thyroid dose increment during time interval i, rem.

ΔA_i = Activity released to environment during time interval i, Curies (Dose equivalent I 131).

$(X/Q)_i$ = Atmospheric dispersion appropriate to time interval i, Ci-sec/m^3-Ci.

B_i = Breathing rate appropriate to time interval i, m^3/sec.
= 3.47×10^{-4} m^3/sec for time = 0-8 hr.
1.75×10^{-4} m^3/sec for time = 8-24 hr.
(average 2.32×10^{-4} m^3/sec for 0-24 hr.

K = Thyroid dose conversion factor for I 131
= 1.49×10^6 rem per Ci-inhaled.