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Exhibit B-4

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

FOURTH SUPPLEMENT TO:

PRELIMINARY SAFETY ANALYSIS REPORT

2180

INTRODUCTION

This amendment contains information on a revised and expanded calculation in answer to Question 4 of the Second Supplement. It also contains a description of the change in the design of the Isolation Valve Seal Water System described in Chapter 5 of the Preliminary Safety Analysis Report and a description of the change in the design of the Safety Injection System described in answer to Question 2 of the Second Supplement.

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4. A potential source of hydrogen following a primary piping failure could be radiolytic decomposition of the safety injection water initiated by the decay energy of the core. Please discuss the magnitude of the gases formed by this process and its potential effect on containment pressure and concentration of free hydrogen in the containment vessel.

Answer

A conservative analysis has been made to estimate the consequences of radiolytic formation of hydrogen. The results of the analysis indicate that an equilibrium condition will be attained in the containment vessel in which the concentration of free hydrogen will be below the lower flammability limit in steam-free air. The analysis and results are based on the following models and assumptions.

The determination of hydrogen radiolysis from the core decay energy following a loss of coolant accident must consider two phases. First, the time during which substantial boiling exists around the core and second, the time after this when essentially no boiling exists.

For the time period in which boiling occurs, the rate of radiolytic hydrogen production would be essentially that associated with the "direct yield" due to radiolysis since under these conditions there is no chance for significant hydrogen concentrations to build up in the boiling water. Under these conditions, a conservative estimate of the amount of hydrogen generated and released to the containment may be made.

Assuming that essentially all of the decay beta energy is absorbed by the fuel, that one half of the decay gamma energy is absorbed by the water surrounding the core and that the formation constant, G_{H2}, for hydrogen is a maximum 0.44 molecules per 100 ev of absorbed energy,[1] the amount of hydrogen formed in the first day after reactor trip is approximately 6 x 10^5 liters of H_2. Assuming that all of this hydrogen is mixed in the containment volume, a free hydrogen concentration of 0.82 volume per cent is obtained.

The emergency core cooling equipment provided for post loss-of-coolant accident conditions will terminate net boiling in the water around the core in approximately the time required to empty the refueling water storage tank which is on the order of 45 minutes to one hour. The amount of hydrogen found during this first hour of boiling is approximately one tenth of the amount produced in one day so the expected concentration of hydrogen due to the presence of boiling in the core is 0.08 volume percent which is well below the lower flammability limit in steam free air of 4 volume per cent. A further case was investigated in which it was assumed that boiling continued for a period of one week following the accident and that all of the hydrogen generated during this period reached the containment. Natural decay of the core gamma source was assumed during this period. The resulting concentration in the containment is 3.3 volume per cent of hydrogen which is below the lower flammability limit of hydrogen in steam free air.

After cessation of boiling, radiolysis of the water continues but a near-steady-state condition is attained where the rate of recombination of radiolytic products equals the rate of dissociation of the water. For this to occur, a certain hydrogen concentration must be sustained in the water phase. We can predict a maximum value for this from published work. Given this value, and utilizing Henry's Law for gas solubility, the hydrogen content of the containment vessel atmosphere may be estimated.

According to Allen[2], gamma radiolysis of static water, results in a hydrogen concentration dependant on the amount of excess oxidant (O_2 + H_2O_2) present in solution. At 25°C, where the oxygen content of air saturated water is a maximum of ∿ 7 ppm for a system at atmospheric presure, the excess oxidant, α, is

$$\alpha = 2\ [O_2]$$

$$= \frac{(2)\ (7 \times 10^{-3} \text{gm } O_2/\text{liter})}{(32\text{gm } O_2/\text{mol})}$$

$$= 4.38 \times 10^{-4} \text{ moles/liter}$$

or,

$$\alpha,\ \frac{\text{moles}}{\text{liter}} = (4.38 \times 10^{-4}\ \frac{\text{mol}}{\ell})\ (6.023 \times 10^{23}\ \frac{\text{molecules}}{\text{mol}})$$

$$= \underline{2.64 \times 10^{20} \text{ molecules/liter}}$$

For this value of α, and utilizing Figure 7.2 in Allen's book, the steady-state hydrogen concentration in solution will be ∿ 1.3 x 10^{19} molec/liter, or

$$[H_2] = \frac{1.3 \times 10^{19} \text{ molec}/\alpha}{6.023 \times 10^{23} \text{ molec/mol}} = 2.16 \times 10^{-5}\ \frac{\text{moles}}{\text{liter}}$$

In order to confirm this, additional calculations were made utilizing Jenk's[3] relationships to give steady-state product concentrations. For this, certain parameters were assumed, as follows (see Appendix):

Volume of Water Around Core = 4000 cu. ft.
Gamma Dose Rate = 0.06 watts/cc
Excess Oxidant in Water, α, 4.38 x 10^{-4} moles/liter
Temperature = 180°F

This computation, given in the Appendix, predicts a hydrogen concentration in the water at steady-state of $\sim 1 \times 10^{-5}$ moles/liter. This is in good agreement with the prediction from Allen.

To demonstrate that the hydrogen concentration is somewhat insensitive to dose rate, the calculations using Jenks' equations were repeated, using a higher dose rate by a factor of 10, i.e., from 0.06 to 0.6 watts/cc. This resulted in a predicted steady-state hydrogen concentration of 2.3×10^{-5} moles/liter, which is also in good agreement with the results from Allen's method.

If we now assume that the net production of hydrogen from gamma radiolysis will occur until the water contains 2.16×10^{-5} moles/liter, and that this solution is in equilibrium with hydrogen in the containment atmosphere, the concentration in the latter may be computed using Henry's Law.

For this, we use the relationship

$$P_{H2} = [H_2] \times K_H$$

where P_{H2} = partial pressure of hydrogen, psia

$[H_2]$ = solution concentration of hydrogen, moles/liter

K_H = Henry's Law constant for hydrogen, psia/(moles/ℓ)

Conservatively, we select K_H[4] at the temperature of minimum hydrogen solubility in water to give the maximum partial pressure for a given concentration in water, i.e., 1.72×10^4 psia/(moles/liter) at 212°F.

Thus, $P_{H2} = (2.16 \times 10^{-5})(1.72 \times 10^4)$

$= 0.372$ psia

and the volume percent in the containment atmosphere will be

$$v/o\ H_2 \cong \frac{0.372 \times 10^2}{14.7}$$

$$\cong 2.53\ v/o$$

The above calculations indicate that the accumulation of hydrogen in the containment following a loss-of-coolant accident due to radiolytic decomposition of water may reach an equilibrium concentration which is about one half of the lower flammability limit of 4 volume per cent.

References

1.) S. Gordon and E. J. Hart, Proc. Second Geneva Conf., 29 13 (1958). Also, R. G. Sowden, J. Nuc. Matl's 8 81-101 (1963).

2.) A.O. Allen, "The Radiation Chemistry of Water and Aqueous Solutions," D. van Horstrand Co. Inc., N.Y. 1961

3.) G.H. Jenks, "Effects of Reactor Operation on HFIR Coolant," ORNL 3848, 1965

4.) D.M. Himmelblau, "Solubilities of Inert Gases in Water" J. Chem. Eng. Data, 5, 10-15, 1960

Appendix

Prediction of Hydrogen Steady-State from Gamma Radiolysis

Reference: ORNL-3848, "Effects of Reactor Operations on HFIR Coolant", G. H. Jenks, 1965

1) Analytic Expressions

$$\frac{k_{14}\ C\ \ \alpha(OH)}{T\ k_9} + \frac{k_{18}\ (OH)}{T\ k_{12}}\ \sqrt{A + \alpha\ k_{12}\ (OH)}\ \sqrt{\frac{k_{14}\ C}{k_{16}\ k_9}} = C-A+B-\frac{k_{14}\ CA}{T\ k_9\ k_{12}}$$

$$T = \frac{C}{k_9} + \frac{2\ (F-C)}{k_8}$$

$$(H_2) = \frac{A}{k_{12}\ (OH)}$$

2) Rates of Formation of Radiolytic Products

The rate of formation of a given product will be given as the product between the dose rate and the G value of the particular radiation involved. For pure gamma radiolysis,

$A = R_{H2}$	and	$G_{H2} = 0.44$
$B = R_{OH}$		$G_{OH} = 2.34$
$C = R_{H2O2}$		$G_{H2O2} = 0.70$
$F = R_{e^-_{aq}}$		$G_{e^-_{aq}} = 2.31$

The dose rate was selected from the average of the decay energy during the first day after reactor trip. That is

$$\text{Dose} = (1.72 \times 10^{22})\ P(t)\ \ \text{Mev}$$

Where P(t) is integrated decay energy in equivalent full-power seconds (efps).

and,

$$A = (1.66 \times 10^{-20})(0.44)(3.75 \times 10^{14}) = 2.74 \times 10^{-6} \frac{\text{moles}}{\ell\text{-sec}}$$

$$B = (1.66 \times 10^{-20})(2.34)(3.75 \times 10^{14}) = 1.46 \times 10^{-5} \quad ''$$

$$C = (1.66 \times 10^{-20})(0.70)(3.75 \times 10^{14}) = 4.35 \times 10^{-6} \quad ''$$

$$F = (1.66 \times 10^{-20})(2.31)(3.75 \times 10^{14}) = 1.44 \times 10^{-5} \quad ''$$

3) Excess Oxidant

As given before, the excess oxidant for air saturated water at 25°C, is

$$\alpha = 2\ [O_2]$$
$$= 4.38 \times 10^{-4} \frac{\text{moles}}{\text{liter}}$$

4) Rate Constants

Rate constants used in the analytical expressions are reported by Jenks for 25°C, and these have been corrected to 180°F for the present case as follows:

$$k = Ae^{-E/RT}$$

and

$$\frac{k_1}{k_2} = e^{-E/R\left(\frac{1}{T_2} - \frac{1}{T_1}\right)}$$

where T_1 = 25°C (298°K) and T_2 = 180°F (355.4°K), we obtain

$$\frac{k_1}{k_2} = e^{-E/R\,(5.3 \times 10^{-4})}$$

During the first day, P(t) was estimated to be 850 efps, but this represents the full amount of decay energy. In this calculation, it is assumed that only 50% of the energy is gamma, and of this, only 50% is absorbed in the water. Thus,

$$\text{Dose Rate} = \frac{(1.72 \times 10^{22})\ (850)}{(4)\ (86{,}400\ \text{sec/day})}$$

$$= 4.25 \times 10^{19}\ \text{Mev/sec}$$

If we assume that this energy is absorbed in 4000 cuft of water surrounding the reactor core, the volumetric dose rate is,

$$\text{VDR} = \frac{(4.25 \times 10^{19}\ \text{Mev/sec})}{(4000\ \text{ft}^3 \times 28.3\ \ell/\text{ft}^3)}$$

$$= 3.75 \times 10^{14}\ \text{Mev}/\ell\text{sec}$$

or,

$$(3.75 \times 10^{14}\ \text{Mev}/\ell\text{-sec})\ (1.6 \times 10^{-13}\ \text{watt-sec/Mev})\ (10^{-3}\ \ell/\text{cc})$$

$$= 0.06\ \text{watts/cc}$$

The rates of formation of radiolytic products, are

$$R\ \left(\frac{\text{moles}}{\ell\text{-sec}}\right) = \frac{G\ \left(\frac{\text{molec}}{100\text{ev}}\right) \times \text{VDR}\ \left(\frac{\text{Mev}}{\ell\text{-sec}}\right) \times 10^4\ \left(\frac{100\text{ev}}{\text{Mev}}\right)}{6.023 \times 10^{23}\ \left(\frac{\text{molec}}{\text{mole}}\right)}$$

$$= 1.66 \times 10^{-20}\ G \times \text{VDR}$$

k	ℓ/mol-sec at 25°C	E, kcal.	E/R	E/R(5.3×10^{-4})	k_1/k_2	ℓ/mol-sec at 180°F
8	1.9×10^{10}	3	1510	0.80	0.45	4.22×10^{10}
9	1.2×10^{10}	3	1510	0.80	0.45	2.67×10^{10}
12	4.5×10^{7}	7	3520	1.87	0.15	3.0×10^{8}
14	4.5×10^{7}	4.5	2260	1.20	0.30	1.5×10^{8}
16	2.7×10^{6}	5	2520	1.34	0.26	10×10^{7}
18	4×10^{9}	3	1510	0.80	0.45	8.9×10^{9}

5) <u>Computations</u>

Substitution of the known or assumed constants into the analytical expressions gives the following

$T = 6.38 \times 10^{-16}$

(OH) 10^{-9}

(H_2) $0.9 \times 10^{-5} \frac{\text{moles}}{\text{liter}}$

$P_{H_2} = 0.155$ psia

v/o H_2 in Containment $\cong$ 1.05%

If the above calculations are repeated, except that the dose rate is increased by a factor of 10, we have

$$\left.\begin{aligned} A &= 2.74 \times 10^{-5}\ \frac{\text{moles}}{\ell\text{-sec}} \\ B &= 1.46 \times 10^{-4}\ \text{"} \\ C &= 4.35 \times 10^{-5}\ \text{"} \\ F &= 1.44 \times 10^{-4}\ \text{"} \end{aligned}\right\}\ 0.6\ \frac{\text{watts}}{\text{cc}},\ 180°\text{F}$$

and the result is,

$$T = 6.38 \times 10^{-15}$$

$$(OH) = 4 \times 10^{-9}$$

$$(H_2) = 2.3 \times 10^{-5}\ \frac{\text{moles}}{\text{liter}}$$

$$P_{H_2} = 0.396\ \text{psia}$$

$$\text{v/o } H_2 \text{ in Containment} \cong 2.69\%$$

ISOLATION VALVE SEAL WATER SYSTEM

A detailed evaluation of this system has resulted in some modifications to the conceptual design described and illustrated in Chapter 5 of the PSAR. These modifications do not alter the principle of leak prevention described in Chapter 5 of the PSAR, but provide several improvements. The modified system is described in this section. Figure 2-1 shows the revised flow diagram of the system.

Principle of Operation

The Isolation Valve Seal Water System functions in the event of any condition requiring containment isolation. When actuated, the Isolation Valve Seal Water System interposes water inside the piping between two isolation points located outside the containment boundary. The water is introduced at a pressure higher than the design pressure of the containment. Seal water tank pressure is maintained by a high pressure nitrogen source which does not require any outside power to maintain seal water pressure. The possibility of leakage from the containment or reactor coolant system past the first isolation point is thus prevented by assuring that if leakage does exist, it will be from the seal system into the containment.

Means of Actuation

Isolation and seal water injection are accomplished automatically in some lines and manually in others, depending on the status of the system being isolated and the potential for leakage in each case. Generally, the criteria for determining which lines are provided with automatic isolation valve seal water injection are as follows:

Automatic seal water injection is provided for lines that could communicate with the reactor coolant system or containment atmosphere following a loss of coolant accident and could be void of water in the event of a loss-of-coolant accident. These lines include:

1. Letdown Line
2. Drain Header
3. Vent Header
4. Pressurizer Relief Tank Vent

Manual actuation of seal water injection is used for lines that are normally filled with water and will remain filled with water following a loss of coolant accident. These lines are routed so that a vertical water leg assures an initial seal on the isolation valve. The manual injection line assures a long term seal.

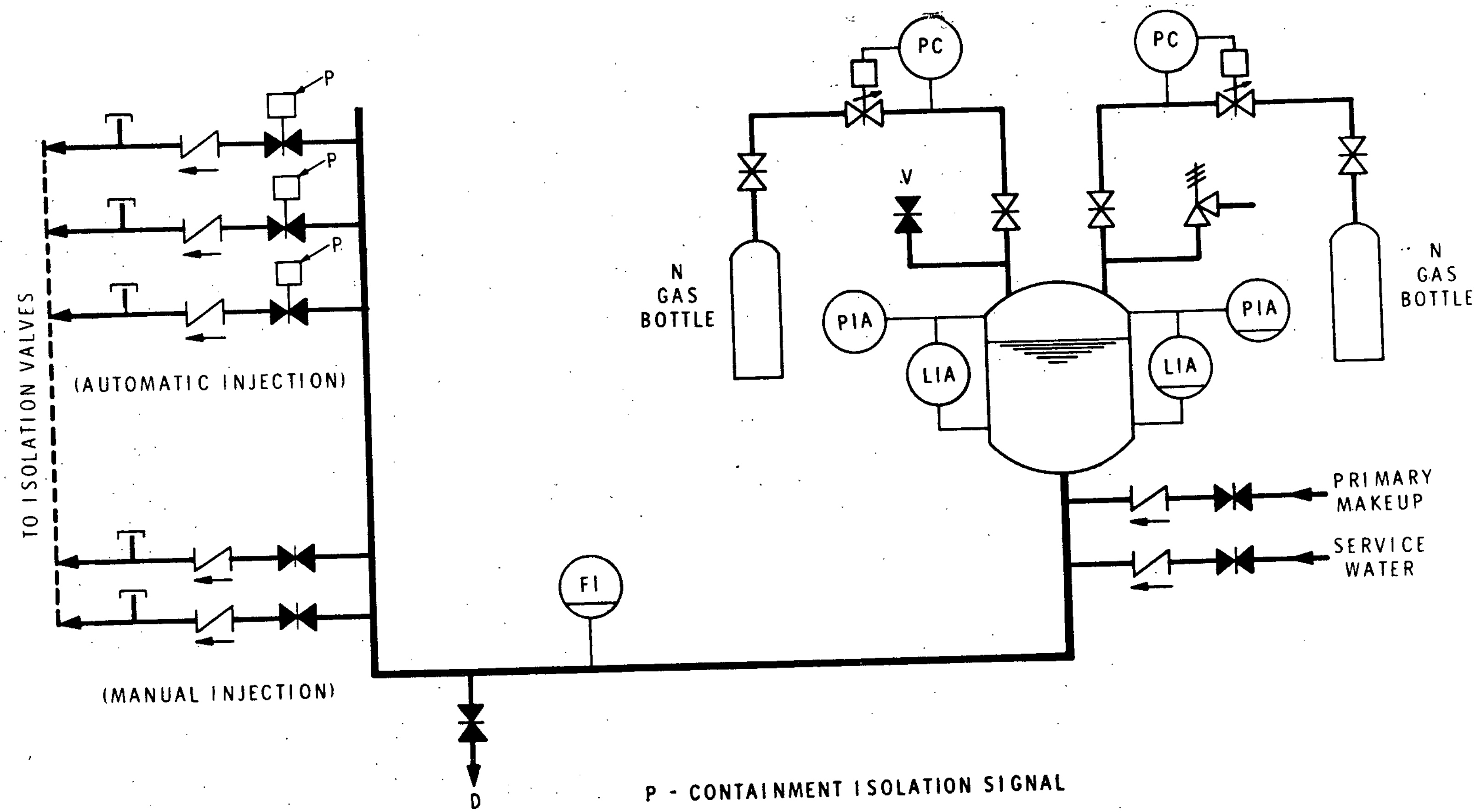
Exceptions to the requirement for seal water injection are: High pressure gas lines, and lines that will always remain full of water because of safety injection service.

High pressure gas lines provide a double barrier to containment leakage without the requirement for seal water injection.

1. Nitrogen Supply Line
   The pressure applied is greater than containment pressure. A pressure regulator maintains the nitrogen pressure and prevents out leakage. A manual diaphragm valve is provided for backup.

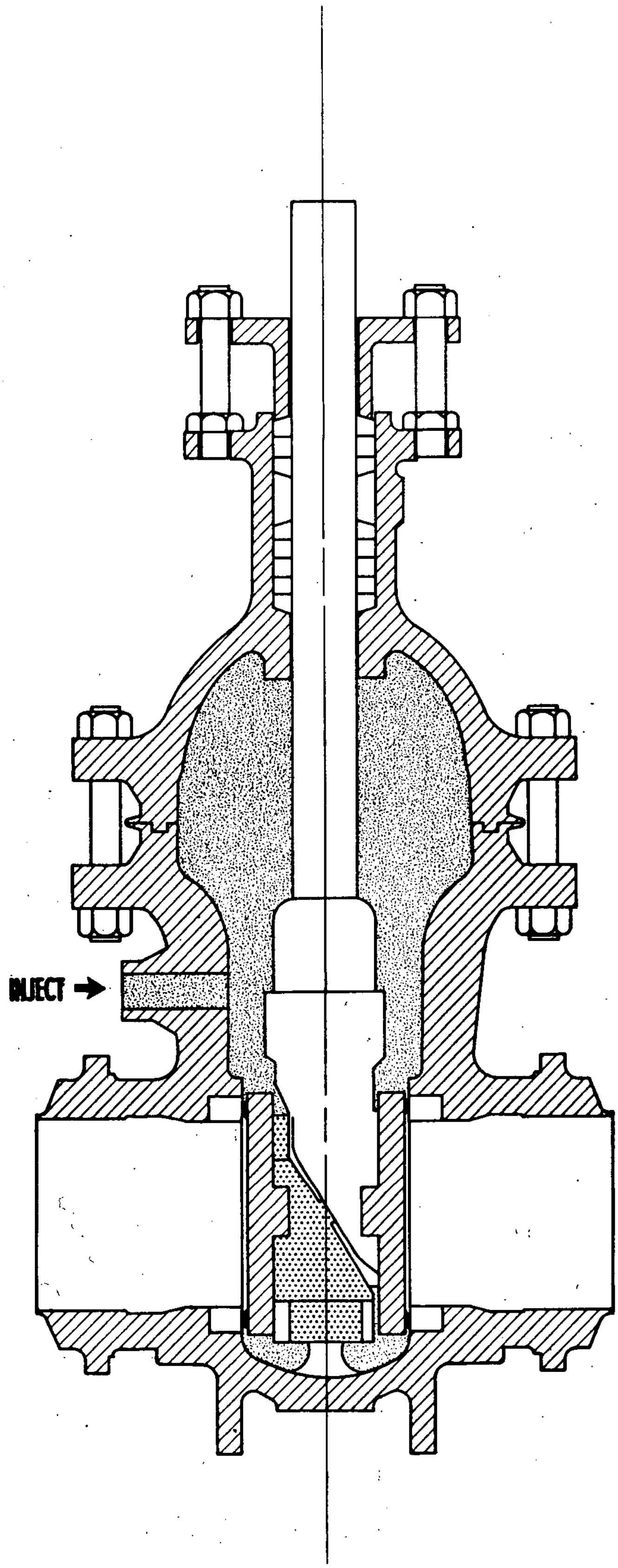
2. Instrument Air Line
   Instrument air pressure is greater than containment pressure. Redundant remote isolation valves close if a loss of pressure occurs. These valves are diaphragm valves to prevent any stem leakage.

Lines in the low head branch of the safety injection system which always remain full for safety injection and recirculation operation do not have seal water injection.



ISOLATION VALVE SEAL WATER SYSTEM

FIGURE 2-1



DOUBLE DISK ISOLATION VALVE WITH SEAL WATER INJECTION

FIGURE 2-2

Both automatic and manual seal water injection schemes add water between the valvesseat and valve packing. This intermediate zone is maintained at a pressure higher than the containment design pressure. There are no real water injection connections between the containment and the first isolation isolation point.

The devices used to provide isolation and a water seal will vary depending on the application. One of the following means will be employed.

For lines approximately two inches and larger, double-disk valves are used for isolation. A drawing of this valve is presented in Fig. 2-2. Double isolation is provided when the valve is closed. The upstream and downstream disks are forced against their respective seats by the closing action of the valve. Seal water is injected through the valve bonnet and pressurizes the two valve disks. The seal water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is provided by two globe valves with the seal water injected into the pipe between the valves. The valves are installed such that the isolation water wets the packing. When the valves are closed for containment isolation, the first isolation point is the valve plug, and the pressurized water seal is applied between the valve plug and packing.

There is a sufficient supply of seal water in the seal water injection tank to assure a one day supply of seal water under the most adverse circumstances. The maximum acceptable leakage in any valve is 10 cc/hr/inch of nominal pipe diameter. Tests on these valves have indicated a much lower leakage can be expected. However, the seal water supply tank is sized for a 24 hour supply, based on the conservative assumptions that all valves are leaking at five times the nominal 10 cc/hr/inch diameter, and each valve is 10 inches in diameter. For the approximately 50 penetrations 600,000 cc's are required. A 200 gallon seal water supply tank meets the requirement. A makeup capability is provided for long term seal water supply.

Should one of the valves fail to seat, flow through the failed valve will be limited to 100 times the nominal leakage value. This is accomplished by an orifice in each seal water injection line. A flow indicator is provided for the system. An operator then has more than one half day to observe the leak and close the injection line or start the makeup supply.

SAFETY INJECTION SYSTEM

The Safety Injection System for the Indian Point Unit No. 2 (as shown in the process flow diagram, Figure 3-1) is made up of several independent subsystems. These subsystems each contain redundancy of equipment and redundancy of flow paths inside of the missile protection boundary. This redundancy assures complete reliability of operation and assures continued core cooling, even in the event of a credible failure of a component.

1. Injection Phase

a) Description

During the injection phase of operation, two independent low-head core deluge systems deliver borated refueling water to the reactor vessel above the core through nozzles on the hot-leg piping in each of the four reactor coolant loops. Either deluge system operating by itself at partial effectiveness with one pump will prevent major damage or melting of the core and will limit the amount of zirconium water reaction to about five per cent for large breaks such as the hypothetical double ended severance of a reactor coolant pipe.

The two independent deluge systems are shown on the process flow diagram, Figure 3-1. One system utilizes the two residual heat removal pumps, each of which take suction from a line from the refueling water storage tank. These pumps deliver borated water through the residual heat removal line, one of the two residual heat exchangers, and through an injection header and four injection lines to the connections on the hot-leg piping of each of the four reactor coolant loops.

The other deluge system utilizes the two low-head safety injection pumps which are located inside the containment. These take suction from a line from the refueling water storage tank, and deliver through the other residual heat exchanger to a separate injection header from which four additional injection lines branch to supply the hot-leg piping connections on each of the four reactor coolant loops.

The two residual heat exchangers are isolated from each other during power operation by locked closed valves in their inlet and outlet headers. The injection lines from the two separate hot leg injection headers utilize the same four injection points on the reactor coolant loops. Each deluge system is designed to fulfill its function of core cooling with one line spilling.

In addition to the two deluge systems, a high head system containing three high-head pumps assures injection of borated water into the reactor coolant system for small breaks where the blowdown is slow and pressure in the reactor coolant system remains high.

These pumps are arranged to deliver to two injection manifolds, each of which supplies two injection lines. A total of four high-head injection lines inject the borated water into the reactor vessel by way of a connection in each loop cold leg. The two injection manifolds and the check valves in the high head safety injection pump discharge header ensure that the full delivery of one high head pump injected even in the event of a rupture of one high head system injection line or failure to deliver flow through one of the two injection manifolds. The high-head system can fulfill its design objective for small breaks even with the failure of one injection manifold to deliver to the reactor coolant system and the failure of any high-head injection pump.

Either of the two low-head deluge systems provide the necessary core cooling without need for the high head system for large breaks including the double ended severance of the reactor coolant pipe.

b) Operation

The safety injection equipment needed during the immediate post-accident injection phase of operation is started automatically upon coincident signals of low reactor coolant pressure and low pressurizer water level from two of three independent pressure and two of three independent level channels. The valves and pumps actuated by the automatic safety injection signal are indicated by the "S" on Figure 3-1.

Equipment in excess of the basic needs for core protection is started automatically. Table 3-1 shows a comparison between the requirements for core protection and the pumping equipment actually placed in operation.

2. Recirculation Phase

Delivery of water to the core is continued for long-term cooling by recirculation of the spilled refueling water and reactor coolant from the containment floor sump by the low head safety injection pumps through the residual heat exchangers to the reactor coolant system.

Recirculation is initiated as the level of water in the refueling water storage tank is reduced below a predetermined point. Injection of refueling water using the high head and residual heat removal low head systems continues during the transfer to recirculation.

The recirculation equipment is essentially as described in Question 2 of Supplement No. 2 to the Preliminary Safety Analysis Report with minor modifications. The low-head safety injection pumps are valved to take suction from the containment sump and recirculate the sump water through a residual heat exchanger to the four deluge injection points on the reactor coolant hot leg piping. A branch connection on the heat exchanger outlet can divert, if required, a portion of the cooled recirculation water for containment spray. One low-head injection pump (already in operation for injection) has sufficient capacity to both spray the containment and to maintain the core in a flooded condition.

TABLE 3-1

Summary of Core Cooling Equipment Operation

Equipment Needed for Core Protection	Equipment Started Outside Electric Power or 3 Diesels Available	Equipment Started Two of Three Diesels*
1 Low-head pump (safety injection or residual heat removal	1 Low-head safety injection pump*	1 Low-head safety injection pump
	1 Residual heat removal pump*	
1 High-head Safety Injection Pump (small breaks only)	3 High-head safety injection pumps	1 Residual heat removal pump
		1 High-head safety injection pump

*Automatic starting sequence starts the alternate pump should one pump fail to start. All valves required to align the systems for injection are moved to their proper position even if power is available from only two of the three emergency diesels.

Thus, for large breaks where there is a potential for fission products release, the recirculation flow path is entirely within the containment, and the injection equipment located outside the containment can be shut down and isolated.

The recirculation loop inside the containment has full-pumping capacity standby in the other low-head safety injection pump. In addition, the residual heat removal pumps, located in the auxiliary building outside the containment, can be placed in service if necessary taking suction from another floor sump inside the containment and recirculating the water through either of the residual heat exchangers to the deluge injection points on the reactor coolant hot leg piping. Each residual heat removal pump has the same head-flow design point as the low head safety injection pumps and is therefore a full-capacity backup. Hence, the installed low head recirculation capacity is four times that required to provide minimum recirculation requirements.

Branch connections to the suction of the high-head safety injection pumps are provided downstream of the residual heat exchangers in each independent low-head system. These connections permit operation of the high-head pumps for recirculation following small reactor coolant system ruptures in which reactor coolant pressure remains high.

The high-head pumps are aligned into the recirculation flow path at the conclusion of the injection phase of operation if the reactor coolant pressure remains above the design delivery head of the low-head pumps. At least one of these high head pumps will remain in operation as the recirculation flow path is aligned. Any one of the three available high head pumps has capacity adequate to maintain the necessary core recirculation flow.

3. Electrical Power Supply

The basic features of the power supply system are as such that with all three of the diesels in service or outside electric power available:

a) Full containment cooling capacity is available using all fan coolers and all spray pumps

b) Double safety injection capacity is available using all high-head pumps and one residual heat removal and one low-head safety injection pump

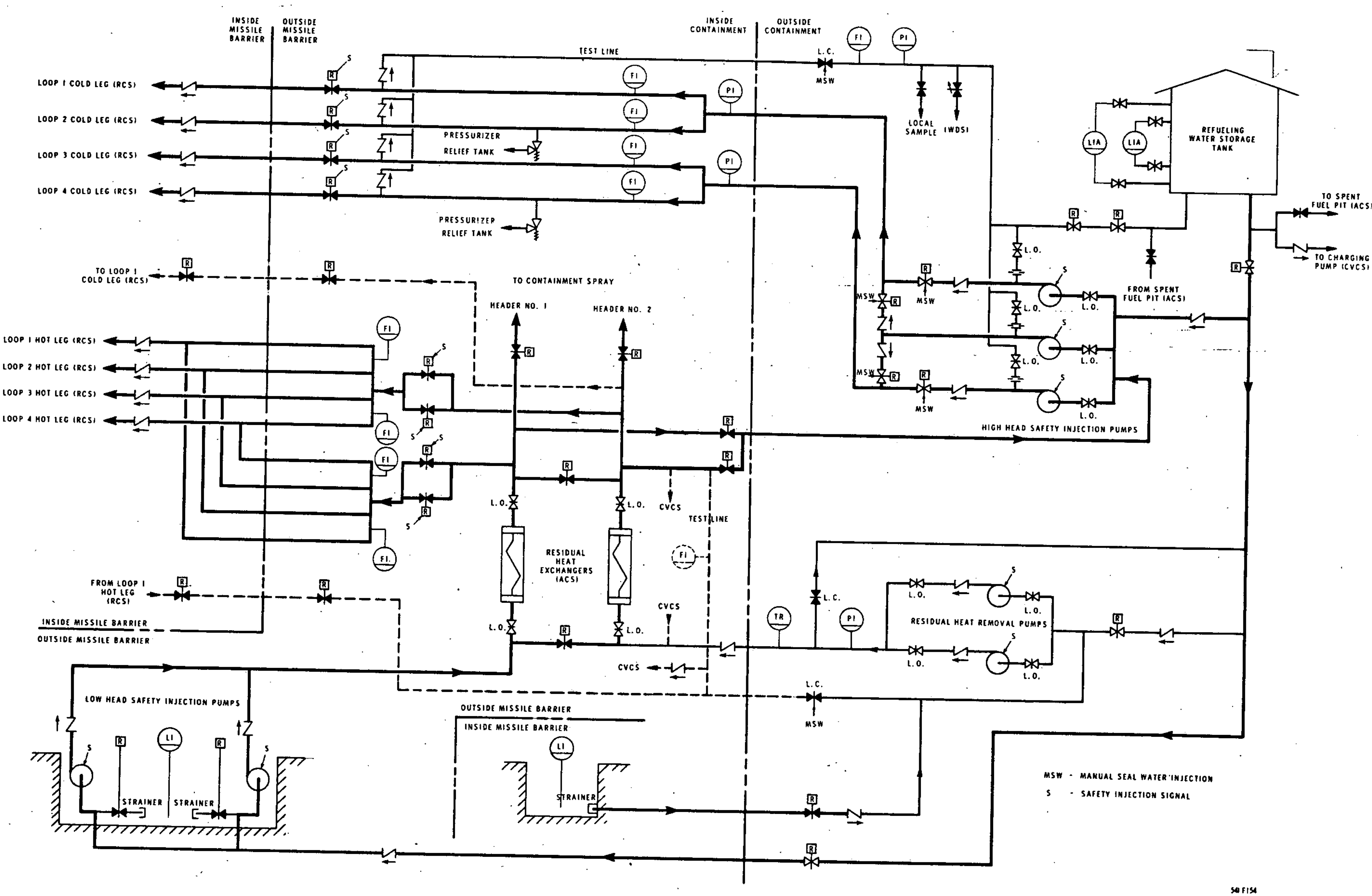
The emergency diesel power supply is sized to provide power with only two out of three units operating to operate necessary engineered safeguards to provide both containment and core cooling following the maximum hypothetical double ended pipe severance loss-of-coolant accident. Table 3-2 lists the engineered safeguards equipment automatically started after a loss-of-coolant accident.

TABLE 3-2

ENGINEERED SAFEGUARDS
IMMEDIATE POST-ACCIDENT OPERATION

Automatic Operation with outside Power or Three Diesels	Automatic Operation with two of Three Diesels	Minimum Requirement*
1 of 2 Residual Heat Removal Pumps	1 Residual Heat Removal Pump	3 Fan Coolers
1 of 2 Low-head S. I. Pumps	1 Low-head S. I. Pump	2 Service Water Pumps
3 of 3 High-head S. I. Pumps	2 High-head S. I. Pump	1 High-head S. I. Pump
2 of 2 Spray Pumps	1 Spray Pump	1 Low-head S. I. Pump
2 of 6 Service Water Pumps	2 Service Water Pumps	
5 of 5 Fan Coolers	3 Fan Coolers	

*The minimum requirement is based upon preventing containment pressure from exceeding the design value.



SAFETY INJECTION SYSTEM
FIGURE 3-1

431 77 501 (4)